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Giardini Naxos

Book of Abstracts

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ITER Construction and Manufacturing Progress Toward First Plasma

Author: Bernard Bigot[1]

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ITER reached in November 2017 completion of 50% of the work required to achieve First Plasma. Progress is most visible in the completion of many key buildings, such as the tokamak assembly building, the cryogenic plant, and the magnet power supply building have been completed. The tokamak building will be ready for equipment in 2020 and the bioshield is already to full height. Key systems begin commissioning in 2018, including the steady-state electric network and the component cooling water, while the cryogenic system and magnet power supply commissioning begins in 2019. Thus, the physical plant is moving rapidly toward completion, and key systems are entering the commissioning phase. Manufacturing progress is equally impressive. The base and lower cylinder of the cryostat have been assembled on the ITER site. Three of the six poloidal field coils and the first of the six modules of the central solenoid are being wound or finished. The winding pack and casing for the first toroidal field magnet are complete and verified to meet the high tolerances required (<0.5 mm). The parts for the first vacuum vessel sector have been fabricated and demonstrated to meet strict tolerances (<1 mm). Therefore, the major components of the tokamak have passed into the fabrication phase. The Heating and Current Drive systems (NB, ECH and ICH) are also in the final design phase. The progression of ITER operation from First Plasma (FP) to the achievement of the Q = 10 and Q = 5 project goals has been formalized in a Staged Approach. This is a stepwise installation of components and auxiliary systems; all systems will be installed before the start of the fusion power operational phase. The ITER Research Plan has been revised in 2017 to be consistent with the systems available in each phase.

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Status European procurement for ITER

Author: Johannes Schwemmer[1]

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Missing

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European roadmap to fusion energy

Author: Tony Donné [1]

[1] *EUROfusion*

The European Roadmap to the realisation of fusion energy breaks the quest for fusion energy into eight missions. For each mission, it reviews the current status of research, identifies open issues, proposes a research and development programme and estimates the required resources. It points out the needs to intensify industrial involvement, to educate the fusion scientists and engineers of the future, and to seek all opportunities for collaboration outside Europe.
A long-term perspective on fusion is mandatory since Europe has a leading position in this field and major expectations have grown in other ITER parties on fusion as a sustainable and secure energy source. China, for example, is launching an aggressive programme aimed at fusion electricity

production well before 2050. Europe can keep the pace only if it focuses its effort and pursues a pragmatic approach to fusion energy. With this objective the present roadmap has been elaborated. The roadmap covers three periods: The short term which roughly covers the period until ITER comes into operation and the DEMO Conceptual Design is completed, the medium term which runs until ITER is in routine operation at high performance and the DEMO Engineering Design is completed and the long term.
ITER is the key facility of the roadmap as it is expected to achieve most of the important milestones on the path to fusion power. Thus, the vast majority of resources proposed in the short term are dedicated to ITER and its accompanying experiments. The medium term is focussed on taking ITER into operation and bringing it to full power, as well as on preparing the construction of a demonstration power plant DEMO, which will for the first time supply fusion electricity to the grid. Building and operating DEMO is the subject of the last roadmap phase: the long term. It might be clear that the Fusion Roadmap is tightly connected to the ITER schedule. A number of key milestones are the first operation of ITER, the start of the DT operation, and reaching the full performance at which the thermal fusion power is 10 times the power put in to the plasma.
DEMO will provide first electricity to the grid. The Engineering Design Activity will start a few years after the first ITER plasma, while the start of the construction phase will be a few years after ITER reaches full performance. In this way ITER can give viable input to the design and development of DEMO. Because the neutron fluence in DEMO will be much higher than in ITER (atoms in the plasma facing components of DEMO will undergo 50-100 displacements during the full operation life time, compared to only 1 displacement in ITER), it is important to develop and validate materials that can handle these very high neutron loads. For the testing of the materials a dedicated 14 MeV neutron source is needed. This DEMO Oriented Neutron Source (DONES) is therefore an important facility to support the fusion roadmap.
The presentation will focus on the strategy behind the fusion roadmap and will describe the major challenges that need to be tackled on the road towards fusion electricity. Encouraging recent results will be given to demonstrate the outcome of the focused approach in European fusion research.
The author is indebted to the whole European fusion community that is together working to make fusion a reality. This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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Role of the Italian DTT in the Power Exhaust implementation strategy

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In the European road map towards the realisation of fusion energy, one of the challenges is the power exhaust for DEMO. If the ITER baseline strategy can't be extrapolated to DEMO, tens of years will delay the realization of a fusion plant. So in parallel to ITER exploitation, it is mandatory to test alternative solutions for the heat loads on the divertor as risk mitigation for DEMO.
In the last years two schemes have been proposed as possible solutions: alternative magnetic configurations and the use of liquid metal divertors. Up to now these solutions have been tested at proof of principle level in devices with a plasma current not exceeding 1 MA and SOL parameters significantly different from reactor conditions. To implement one of these concepts on DEMO it is necessary to make another intermediate step, otherwise the extrapolation to DEMO is too large by upgrading existing facilities or by building a dedicated Divertor Tokamak Test (DTT) facility.
In this framework, the Italian fusion community with the involvement of European Labs has proposed a new device, the Italian DTT, in which one or more alternative magnetic configurations or/and liquid metals can be tested in DEMO relevant conditions.
To fulfil the DEMO requirements the device have to operate at relevant heat load on the divertor maintaining high core performance, i.e. operations in an integrate scenario with core and edge parameters as close as possible to ITER and DEMO.
The Italian DTT project has fully supported by the Italian Government that has also identified and is implementing a funding scheme.
DTT will be a high field superconducting toroidal device (6 T) carrying plasma current up to 6 MA

in pulses with length up to 100s, with an up-down symmetrical D-shape defined by major radius R=2.15 m, minor radius a=0.7 m, and an elongation around 1.7.
The status of the project will be illustrated, highlighting the reviewed design addressed to increase the flexibility by allowing for fully double null operation.
The site, the time schedule, and the cost estimation will be presented too.

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Optimal design of DMA probe for austenitic stainless steel weld of CFETR vacuum vessel

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As to the ultrasonic testing of argon arc seam of 50mm austenitic stainless steel China Fusion Engineering Test Reactor(CFETR) vacuum vessel mock-ups, there are some limitations if we adopt the traditional ultrasonic probe or linear array phased array probe. In this paper, we designed a Dual Matrix Array(DMA) probe based on the CIVA, and then analyze the optimal principle of the probe parameters. The results show that the DMA probe's signal to noise ratio much higher compared to the traditional probe, and the surface blind area is reduced . The defect detection rate meets the requirements of relevant standards. The research result has much reference value for the application of phased array ultrasonic testing in austenitic stainless steel welding joint.

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Characteristics and Experiment Measurement of Cascaded Plasma In Linear Plasma Devices

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Finite numerical simulation on plasma generation, confinement and distribution, is lacked in design and research of linear plasma devices. As a high density plasma source, cascaded arc plasma is widely used in plasma and material interaction devices. In this paper, a high-density linear plasma device with cascaded arc source is developed, which plasma parameters and distribution is analyzed by COMSOL Multiphysics. The simulation results show that for argon arc discharge, with magnetic field 2000Gs on axis, argon gas flow 100cm3/s, 80A between cathode and anode, plasma density with distance of 200mm from anode is 1.42×1022m-3, and the election temperature is 1.15eV. To validate the model, results are compared with the experimental findings, in which Langmuir probe is adapted to discover plasma parameters, agreed with the numerical simulation well. Research result can provide references for engineering design.

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Decontamination tests of dust under load for the ITER blanket remote handling system

Co-author: Makiko Saito [1]

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Radioactive dust will accumulate in the vacuum vessel (VV) of ITER after plasma operations. Thus, the ITER Blanket Remote Handling System (BRHS) will be installed in the VV to handle the blanket modules, which can weigh up to 4.5 ton and be larger than 1.5 m, stably and with a high degree of positioning accuracy. The BRHS itself also needs to undergo regular maintenance in the Hot Cell Facility (HCF). Maintenance workers will be exposed to the radioactive dust that adheres to the surface of the BRHS. Past studies estimated the contaminated surface area of the BRHS, however, in this study, decontamination tests were performed and the dose rate to maintenance workers was calculated using the Monte Carlo N–Particle Transport Code (MCNP5). Decontamination tests were performed by using multiple test pieces of varying surface roughness (Ra 12.5, Ra 6.3, and Ra 1.6) made from SUS329J4L and two different types of brushes to simulate decontamination of the BRHS surface. Tungsten dust was pressed on the test pieces to simulate the loading by the rollers. After the test pieces were brushed the surface of each test piece was observed by using both optical and scanning electron microscopes. Dust was reduced by approximately 99% in all cases where the SUS304 brush was used regardless of surface roughness. Afterwards, the decontamination rate (amount of dust that was cleaned) was used to estimate how much dust will be able to be cleaned from the BRHS surface. This paper describes the optimal decontamination tools with respect to the BRHS surface conditions to calculate and hopefully reduce the dose rate to maintenance workers so as to optimize the BRHS maintenance plan in the HCF.

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First version of the W7-X Fast Interlock System

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The central safety system (cSS) of W7-X consists of two parts. The safety related PLC with its corresponding periphery, such as sensors and actors, fulfills the requirements of occupational safety and ensures basic investment protection. The reaction time from the signalization of dangerous faults to the initiation of protection measures like W7-X emergency stop or media shut-off is in the range of some 100 ms. Because this is not sufficient to protect components in the plasma vessel in case of overloading by plasma heating, a fast interlock system was implemented. In the first operation campaigns of W7-X, with limited energy input, only a local fast interlock of the ECRH (using stray radiation probes) has been applied. Now, the central fast interlock system (cFIS) as part of the cSS and local fast interlock systems (lFIS) in several heating systems have been installed. Together with a set of safety relevant diagnostics, this system is able to react to dangerous situations within the few ms-range.
The paper describes the system design, hardware components as well as the software implementation in detail. First results, which are obtained during the operation campaign 1.2b in summer 2018, are outlined.

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

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Thermal mixing enhancement of liquid metal film-flow by various obstacles under vertical magnetic field

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Heat removal from liquid metal film flow has been widely studied for liquid divertor concepts of fusion reactor. In this study, thermal mixing characteristics of the liquid metal film-flow with locally heated on the surface under the vertical magnetic field was experimentally investigated by using various types of obstacle as a vortex generator. The temperature distributions on the bottom wall were measured by 40 thermocouples installed on the channel bottom downstream of the vortex generator. In order to evaluate how much heat transports from the locally heated free-surface to the bottom wall, the efficiency of heat transport was investigated for the various vortex shaped generator. The velocity distributions were measured with using a rake consisted of 40 electrical potential probes on the downstream of the channel. A delta-wing, hemisphere or cubic obstacle was used as a vortex generator installed at the center of the bottom. The experiments were conducted for laminar flow region where Re=1000-1700 and in the range of N(=Ha2/Re)=0-36.0 in the presence of the vertical magnetic field in the acrylic rectangular duct. GaInSn alloy was used as a working fluid. According to the comparison of heat flux distributions obtained by the experiments, the entire distributions moved towards the upstream with increasing of the strength of the vertical magnetic field for all the vortex generator types. This tendency meant the heat removal from the free-surface of liquid metal film-flow quickly in case of relatively high vertical magnetic field. The horizontal velocity distribution of the liquid metal film-flow obtained by the probe rake showed the typical M-shape distributions in case of without the vortex generator, some MHD characteristics of the velocity fluctuations were observed in case of various types vortex generators.

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Ultra high vacuum ZnSe window flange design for Phase Contrast Imaging diagnostics for the Wendelstein 7-X stellarator

Co-authors: Christoph von Sehren [1]; Olaf Grulke [2]; Ralph Laube [1]; Adrian von Stechow [1]

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The Phase-Contrast Imaging (PCI) system is used to measure plasma density fluctuations in the W7-X stellarator at the Max-Planck-Institut für Plasmaphysik (IPP) in Greifswald, Germany. For this purpose, an expanded CO2 laser beam with a wavelength of 10.6⊠m passes through the plasma and the scattered laser beam components yield information on plasma density fluctuations. The laser beam is expanded on an optical table by telescope optics to the desired beam diameter. A mirror system directs the beam through a zinc selenide (ZnSe) window at the entry port flange directly through the plasma center. Through a second opposite ZnSe window on the exit port flange, the beam is directed via a similar set of mirrors to the receiver optical table where the measurement takes place. A typical diameter of the laser beam is approx. 120mm. Due to the wavelength of the laser beam, ZnSe windows must be used.
The challenge is to seal the window, which represents the vacuum barrier, into the counter flange ultra high vacuum (UHV) tight. Additionally, the seal needs to fulfill W7-X requirements as compatibility with the bake-out temperature of 150℃ and electronen cyclotron resonance heating (ECRH) radiation resistance.
A standard solution is not provided commercially, since ECRH resistance demands for an all-metal seal and VITON® or another elastomer seal cannot be used. The challenge is the special window material (ZnSe) and the large window diameter. A metal gasket based on the principle of the CF gasket considered as a safe standard connection in UHV technology, but couldn't be used on the glass side. One solution seemed to be the HELICOFLEX® seal, which combines the elastic component with the temperature resistance. In addition to the special sealing concept for the PCI, the poster shows a market overview of the standard windows for the UHV area.

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Tungsten coatings repair: an approach to increase the lifetime of plasma facing components

Co-authors: Mathilde Diez [1]; Mehdi Firdaouss [1]; Marianne Richou [2]; Helene Roche [1]; Marc Missirlian [1]

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Tungsten coatings have received a great deal of attention as a technical solution for plasma facing components (PFC) in present-day tokamaks owing to their advantages over bulk tungsten, such as lower cost and weight. Nevertheless, tungsten (W) coatings are hard and fragile. Their lifetime is mainly limited by two degradation mechanisms occurring during the operation of the tokamak: erosion resulting from exposure to long plasma discharges and local damages caused by runaway electrons.
In this paper, an alternative solution to the current replacement of damaged PFCs is proposed. It consists of combining a stripping method and a physical vapor deposition (PVD) method for repairing their surface, with a particular interest for actively water-cooled components made of copper alloy.
First, the process of removing tungsten was investigated on CuCrZr samples coated with a 15µm layer of W using 3 carefully selected techniques: laser ablation, sandblasting and chemical attack. For each technique, etching properties were assessed by means of microscopic observations and topography analyses. The results presented here also highlight the issues related with controlling the etching depth and the treated surface area. The second part of the paper focuses on the fabrication of a new W coating layer by PVD. To ensure the performance of the coating after repair, the properties of the new W layer were characterized and compared to the one of the old coating.
In addition to establishing and validating a whole repair procedure for PFCs, this study will provide technical specifications to consider the integration of this technology on a robotic arm. A first application could be the in-situ repair of W coatings in WEST tokamak. From a technical point of view, the integration of different systems on a robotic arm might also be beneficial for ITER where maintenance and inspection will be carried out by robots.

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Development of reliable tooling and processes for remote maintenance of ITER cooling water connections

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The high neutron flux inherent in fusion reactors creates high heat loads in the components surrounding the plasma. These heat load needs to be managed through active cooling. These components also become highly activated so require remote maintenance, hence the connection and disconnection of these cooling systems becomes an important functionality of these maintenance activities. The integrity of these connections is also of critical importance both to the operation of the components and to maintain the confinement of the activated effluent they contain; therefore, it is imperative that a reliable system of connection and disconnection is established.
TIG welding is a process that is recognised within nuclear design standards as a reliable technique that can be used for connections on confinement barriers. RACE has performed extensive work across a range of applications in ITER such as the diverter, diagnostic port plugs and neutral beam vessel, and pipe sizes varying from DIN 25 to DIN 200 to develop tooling and processes to ensure that high quality TIG welding can be conducted remotely.
This work shows that autogenous TIG welding is an appropriate technology to conduct this maintenance activity. It also notes that for this technology the pipe thickness joint thickness must be restricted to a maximum of 3mm and that the sulphur content of the stainless-steel material must be closely controlled with smaller tolerances that anticipated by the industrial codes. The use of filler inserts to supplement material is also discussed. It concludes that the control of the alignment

of the tooling is critical to success and suggests design solutions that can achieve these requirements.

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Electromagnetic modelling and design of DEMO and disruption location prediction

Co-authors: Francesco Maviglia [1]; Raffaele Albanese [2]; Roberto Ambrosino [3]; Gianfranco Federici [1]; Massimiliano Mattei [4]

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The design study of a DEMOnstration (DEMO) Fusion Plant is one of the main points of the European Roadmap to Fusion Electricity [F. Romanelli, http://www.efda.org/wpcms/wp-content/uploads/2013/01/JG12.356-web.pdf]. The pre-conceptual design phase of DEMO is presently used to explore a flexible range of the main machine geometrical design parameters, including machine magnetic configurations, and optimisation of plasma scenarios and conductive geometries.
In this paper is presented the electromagnetic modelling of the DEMO baseline scenario, including the analysis and design activities on the plasma surrounding electrically conductive structures, and their influence on the passive vertical stabilisation (VS), which is a machine size driver, due to the large amount of power needed to vertically stabilise the plasma, and the limitations on the poloidal field coils. A careful design of the blanket first wall poloidal geometry was performed, taking into account the plasma heat load on the wall, allowing the minimisation of the distance between the plasma and the vacuum vessel, which is the closest electrically conductive toroidally continues structure . Both the improvements on the passive and active [R.Ambrosino, this conference] VS allowed to increase the maximum controllable plasma elongation at 95% of the separatrix, from 1.59 to 1.65, from the previous to the most recent DEMO baseline, corresponding to an increase on fusion performances.
Finally a study was carried out on the possibility to predict the plasma final position, following a vertical displacement event, which is essential for a DEMO wall protection strategy from plasma transients. Such point is located where the field after disruption is tangential to the FW and pushing the plasma onto it, under the assumptions that the current quench time is faster than the L/R time constant of the passive structures. The final position was also verified using dynamic simulations and images from JET fast camera during a VDE.

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Digital valve system for ITER remote handling – design and prototype testing

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ITER-RH system is used to exchange the divertor's 54 cassette assemblies in the vessel. Water hydraulics and servo valves are currently used in the task requiring high accuracy tracking and the use of de-mineralized water. The main concern has been robustness of the technology. Only few suitable commercial water servo valves exist and problems e.g. with jamming and wear been encountered. A possible mitigation is to use redundant valves but ensuring required level of water cleanliness is still an issue.

An alternative option is to use digital technology where on/off valves and intelligent control is used to produce proportional output. So far, no proper high-pressure on/off valves existed in the market and therefore a new concept was developed and tested with promising results. The valve has very fast response time, flow capacity that suits well for the required velocities and is compatible with de-mineralized water. In addition, the valve is not sensitive against water cleanliness and can be made rad-hard when necessary.

A mock-up of the remote handling system was used as test bench. The system was simulated in order to dimension the valve system and to tune the valve controller. A complete valve package with 16 prototype on/off valves and a manifold was manufactured and assembled. The rest of the components were off-the-shelf and the result is fully compatible to be used as direct replacement for the servo valve system.

Long-term tests of 60 working days and 2000+ hours was made. A new level of performance was demonstrated throughout the tests as tracking accuracies were approximately ten times better than with servo valve. Although some design faults where detected and system had faulty components, tracking accuracy was very good. Tracking and positioning capability of digital valve system exceeds all requirements and seems applicable for ITER-RH application.

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WEST CODAC software quality management

Co-authors: Julian Colnel [1]; Sylvain Bremond [1]; Gilles Caulier [1]; Philippe Moreau [1]; Benjamin Santraine [1]; Nathalie Ravenel [1]

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COntrol, Data Acquisition and Communication (CODAC) real-time software codes are key elements for the operation of a fusion device as they can play a key role both for the machine protection and for the optimization of the experiments. The updating or upgrading of these software codes may be needed quite frequently in order to either correct bugs or include new functionalities, while these code modifications may introduce new errors or bugs that may have an impact on machine operational availability or even on some machine protection items.
A new software quality management process has been developed and implemented in the WEST (W -for tungsten- Environment Steady-state Tokamak) CODAC in order to tackle these issues. Each source code modification, defined by a unique identifier called "production tag version", is logged by the corresponding responsible officer, reported to the relevant team and finally submitted to validation by the relevant team leader. The scheduling of the deployment of any new version in the WEST CODAC package is coordinated at the appropriate level in order to minimize the possible impact on operation.
The "production tag version" of each software code is also archived for each pulse as well as the pulse parameters, which allows coming back to a previous version whenever needed.
The paper will first describe in details the principles used for the new WEST CODAC software quality management. The workflow process and associated tools will then be discussed. A few examples will eventually be detailed.

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KSTAR Tokamak Visible Image sequence classification with Long-term Recurrent Convolutional Networks

Co-authors: Giil Kwon [1]; Hanmin Wi [2]; Jasic Hong [1]

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This paper represents the tokamak in-vessel image sequence classification method that used to automatically infer plasma status. Fast framing standard CCD cameras are installed on KSTAR (Korea Superconducting Tokamak Advanced Research) to monitor plasma shape, plasma motion and plasma status. The images generated by the CCD cameras were used for plasma start-up studies and plasma disruption studies. In KSTAR, the images generated by the CCD camera are visually confirmed after the experiment or analyzed manually by the researcher. We introduced long-term recurrent convolutional networks to automatically infer the plasma status from the image. To obtain the description about the image from cameras, we applied Convolutional neural networks(CNNs). To learn a description of image sequences, description from CNN are used as input to Long Short-Term Memory(LSTM) modules. By applying LSTM, we can learn temporal information from variable-length input image sequence. Our results show this model can effectively learn tokamak in-vessel image sequence and infer status of plasma.

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On Structural Analyses of the ITER Vacuum Vessel Bolometer Camera Housing Conceptual Design

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The ITER bolometer provides an absolutely calibrated measurement of the radiation emitted by the plasma which is a part of the total energy balance. The development is especially challenging because of the extreme environmental conditions within the vacuum vessel (VV) during plasma operation. The bolometer has to guarantee reliable measurements within an environment characterized by high neutron flux as well as temperatures exceeding 200 ℃. In addition to the thermal loads the bolometer body is exposed to the mechanical loads caused by electromagnetic forces during transient events called disruption.
This paper describes a possible procedure for a structural analysis of the bolometer camera body. To examine all important structural properties of the bolometer body, a multiple nonlinear finite element model based on a CAD conceptual design, has been generated. Subsequently, a transient mechanical analysis has been performed using the finite element code ANSYS. The input for the analyses was generated by a general electromagnetic model, taking into account the contribution of all structural parts and electromagnetic loading starting with the DINA code. From the wide range of DINA results the worst case load scenario has been chosen.
This analysis enables the study of the response of the bolometer camera structure to the dynamic excitation caused by electromagnetic forces during the plasma disruption. Moreover, the influence of the different design solutions and different material properties has been investigated. Due to dynamic oscillating excitation by electromagnetic forces the whole bolometer structure is thoroughly shaken, as demonstrated by the results of deformation analyses. The analyses results will be used to validate the design against the required structural integrity during worst-case scenarios and verify the reliability of the bolometer camera design during operation. Finally, this analysis provides the results needed to perform the fatigue analysis of the VV bosses supporting the bolometer camera.

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Servo valve endurance test for Water-Hydraulic systems in ITER-relevant conditions

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ITER Divertor maintenance equipment work under considerable ambient temperature and radiation load. The heavy components are moved with equipment powered with water hydraulics, with demineralised water as a pressure medium. None of this has yet been tested in ITER-relevant environmental conditions and over projected duty cycles and loading. Hence, a project was undertaken to ascertain the component compatibility with the environment and pressure medium, and their robustness over the required operational period. No irradiation, however, was included. This would be done later on if this test was successful. A heated chamber was constructed to emulate the 50℃ ambient temperature of the divertor area. All the tested components were located within this chamber.
Test trajectories and loads were derived from the Divertor Test Platform 2 (DTP2) at Tampere, Finland, at which the Cassette Multifunctional Mover (CMM) operations and equipment are prototyped. The prototyped operations are along the operations projected for the ITER Divertor area maintenance operations with active-to-idle ratio similar to what was projected for ITER operations. The most important component to be tested was the servovalve. As the hydraulic medium is rather aggressive and the selected servovalve was not originally designed to be used with water, the 2000-hour operational time was considered to be a potential issue. Hence, test routines and measurements were specifically tailored to measure the operational parameters of the servovalve.
As a result, after the 2000-hour test the servovalve parameters remained within the limits promised by the supplier (Moog hydraulics). Null-point leakage increased by 45% (from 0.45lpm to 0.65lpm which is still quite low) and tare leakage did not change significantly. Pressure gain and hysteresis increased significantly but remained within the allowable limits. Cylinder tracking error remained more or less constant over the test, although it decreased an insignificant amount over the last testing week.

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X-ray induced defects in advanced lithium orthosilicate pebbles with additions of lithium metatitanate

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Advanced lithium orthosilicate (OSi) pebbles with additions of lithium metatitanate (MTi) as a secondary phase have attracted international attention as an alternative candidate for the tritium breeding in nuclear fusion reactors. In this research, the formation of radiation-induced defects (RD) in the OSi pebbles with various contents of MTi was analysed using X-ray induced luminescence (XRL). After XRL measurements, the accumulated RD were investigate using electron spin resonance (ESR), thermally stimulated luminescence (TSL) and diffuse reflectance spectrometry.
The advanced OSi pebbles with additions of MTi are biphasic without solid solutions, and thus the formation mechanism and the structure of the formed RD during irradiation with X-rays is similar to the single-phase materials. In the XRL spectra, several bands with maxima at around 380, 420 and 810 nm were detected. The band at around 420 nm can be attributed to the formation of E' centres (≡Si·) in the OSi phase, while remaining maxima are the result of the additions of MTi. In the ESR spectra, at least three first derivative ESR signals with g-factors 1.93, 2.003 and 2.040 were detected and attributed to Ti^{3+} centres, E' centres and oxygen related defects. The TSL glow curves are complex and consist of several overlapped peaks between 300 and 450 K, while the TSL spectra consist of one main band with a maximum close to 450 nm. The blue emission was attributed to the radiative recombination of E' centres or some variants of oxygen deficiency centres with oxygen related defects. It is assumed that the additions of MTi can increase the probability for the recombination processes of primary RD in the advanced OSi pebbles during irradiation and thus reduce the formation of chemically stable radiolysis products, for example colloidal lithium particles, which can interact with the generated tritium and form thermally stable lithium tritide.

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Design evolution of the diamond window unit for the ITER EC H&CD upper launcher

Co-authors: Gaetano Aiello [1]; Natalia Casal [2]; Mario Gagliardi [3]; Timothy Goodman [4]; Mark Henderson [2]; Andreas Meier [1]; Gabriella Saibene [3]; Theo Scherer [1]; Sabine Schreck [1]; Dirk Strauss [5]

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The torus window unit is a very particular component of the ITER EC H&CD upper launcher aiming to provide the vacuum and confinement primary boundary between the vacuum vessel and the transmission lines (TLs). The high power 170 GHz millimeter-wave beams generated by the gyrotrons travel along the TLs and pass through the window units, before being quasi-optically guided into the plasma via the upper launchers. The design of the window unit shall thus meet stringent requirements to guarantee the safety function, the millimeter-wave beam transmission and the structural integrity during normal operation and off-normal events. The unit consists of an ultra-low loss CVD diamond disk brazed to two copper cuffs; this structure is then integrated into a metallic housing by welding. The compliance with the requirements shall be assured by applying the ASME Section III – Subsection NC code and a dedicated experimental qualification program.
This paper reports the way in which the design of the unit, already optimized by FEM analyses against the ITER loading conditions, was further improved by the application of the ASME III-NC code, leading to a more feasible and simpler manufacturing and assembling sequence. In addition, the impact of the ITER project decision to change the inner diameter of the waveguide from 63.5 to 50 mm, to improve the beams' mode purity, was assessed and it is also discussed. Different materials for the metallic housing and in particular for the millimeter-wave inserts of the unit were compared using appropriate engineering criteria to mitigate the significant increase of the millimeter-wave thermal loads on the waveguides when the diameter is decreased.

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Prebaking of T-15MD vacuum vessel

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Presently, the Tokamak T-15MD is being built in the NRC "Kurchatov Institute". Vacuum vessel was manufactured and passed the preliminary vacuum tests at the plant in St. Petersburg (Efremov Institute) in 2016. Vacuum vessel consists of toroidal shell made of a 321stainless steel of 5 mm and 8 mm thick, horizontal and vertical ports (152 in total), in-vessel elements. The chamber volume is 47 m3 and a surface square faced to plasma is ~ 200 m2. The purpose of vacuum vessel prebaking is the checking of quality of the numerous welds and a workability of the control system. To bake the vacuum vessel up to 220℃ at the plant in Bryansk, the ohmic heaters (a single 1.2-mm diam. Ni-Cr alloy wire, housed inside a stainless-steel 6-mm diam. shell, with a magnesium oxide ceramic insulator) have been laid on vessel shell surface both outside and inside. The thermal insulation (cases with mineral wool) closed the vessel surface outside. The surface temperature is controlled by thermocouples. The currents through the heaters are regulated by means of control system. The temperature data processed and stored by means of the data acquisition system. The results of vacuum vessel baking are presented.

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Metrology for integration and installation activities at the PRIMA Test Facility

Co-authors: Samuele Dal Bello [1]; Andrea Barzon [1]; Fabio Degli Agostini [1]; Daniele Fasolo [1]; Luca Franchin [1]; Marco Tollin [1]; Alessandro Lo Bue [2]; Luigi Semeraro [2]

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The ITER project requires at least two Neutral Beam Injectors, each accelerating up to 1MV a 40A beam of negative deuterium ions, so as to deliver to the plasma a power of about 33 MW for one hour.
Since these requirements have never been experimentally met, it was recognized necessary to build-up a test facility, named PRIMA, that is in an advanced state of realization and which includes both a full-size negative ion source (SPIDER) and a prototype of the whole ITER injector (MITICA).
The paper describes the main metrology activities performed in the last three years devoted to the integration and installation of the large number of items and plant units composing the facility. Particular emphasis is given to the propaedeutic activities consisting mainly in the definition of the metrology network (the so called Unified Spatial Metrology Network - USMN) by using technologically advanced laser trackers. The USMN is a feature of the Spatial Analyzer software (SA) compliant with the ISO standard that using the motecarlo method is capable to reduce the global measurement uncertainty. The method is based on the installation and measurement of a large number of fiducial points (approximately two hundreds targets). For PRIMA, some local USMN networks have been built up at different locations resulting eventually in the definition of the PRIMA USMN network. This approach allowed the definition of the global reference frame to be used for the positioning of all items, while respecting the uncertainty requirements of each component. In the paper some instances will be given like the positioning of the transmission line (uncertainty of 0.2mm over more than 100m between first and last tank) and the high voltage bushing support structure for MITICA, the vacuum vessel and the beam source for SPIDER (uncertainty better than 0.01mm of the grid apertures).

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Critical Design Issues in DEMO and Solution Strategies

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The EU fusion roadmap defines as a goal the development of a DEMO, which achieves a high plasma operation time and demonstrates Tritium self-sufficiency and net electricity output. A number of design issues have been identified as critical, either because the solution chosen in ITER is not suitable in DEMO or because it is a DEMO-specific issue not present in ITER. All of these will affect in their resolution the design and possibly the technology of several tokamak and plant systems or even the DEMO architecture: (i) Feasibility of wall protection limiters during plasma transients, (ii) integrated design of breeding blanket and ancillary systems, (iii) power exhaust taking advantage of advanced divertor configurations, (iv) tokamak architecture based on vertical blanket segments, (v) direct or indirect power conversion concept, (vi) configuration of plant systems in the tokamak building, (vii) feasibility of hydrogen separation in the torus vacuum pump and direct recirculation, and (viii) plasma scenario.
For each of these issues potential solutions have been identified and activity plans have been defined for the associated developments and assessments. Four of these particularly affect the integrated design of DEMO, namely (i), (ii), (iv), and (vi). These will be introduced and discussed in this article and for each a summary of the identified risks, the rationale for the chosen solution concepts, and the identified required verifications will be given.

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Analysis of inner divertor materials of JET C-wall and ILW from viewpoint of spectrometric investigations

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Installation of metallic plasma facing wall (ITER-like wall – ILW) [1] and replacing the previous carbon wall (JET-C) in the Joint European torus (JET) was a unique possibility to collect from the tokamak vacuum vessel the first wall erosion products (EP) – dust and flakes.
Fundamental investment about the properties of EP to comply with security reasons is given by analysing EP from other tokamak devices. Carbon based materials are considered as plasma facing materials in stellarators. A comparison between tritium release and chemical composition of plasma exposed materials will allow to expand the knowledge about materials behaviour in fusion devices.
Temperature programmed tritium thermodesorption results show to differences between ILW and C-wall plasma facing surface samples. Tritium release from a sample, cut from ILW inner divertor vertical tile, is in range 470-870 K. From analogous position from JET-C wall tritium releases 450-1180 K, while from EP: 370-1140 [2].
Selected EP were investigated with means of energy dispersion X-ray (EDX), infrared, electron spin resonance and Raman spectrometry.
EDX analysis of EP shows presence of metallic impurities (Fe, Ni, W etc.) and carbon as main component. With electron spin resonance spectrometry two types of paramagnetic centres - g=2.002 and g=2.12, are characterized. Raman spectra allowed to estimate that in EP are graphite nano-crystals with size ~15 nm. Infrared spectra show presence of inorganic oxides. The obtained results supplement the information about composition of the EP from fusion devices.

1.M.Rubel et al./Nuclear Fusion 57 (2017) 066027
2.L.Avotina et al./Advanced Materials and Technologies, Palanga, Lithuania, 2015, 144, P125

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Design and analysis of robot for the maintenance of divertor in DEMO fusion reactor

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DEMO represents DEMOnstration Power station, which is a nuclear fusion power station and it is proposed to be built after ITER experimental nuclear fusion reactor. It is impossible to do any change or repair work during the nuclear operation by human directly due to the high radiation and extreme temperature in the fusion reactor. And the solution to solve these issues is adopt the remote handling robot.
Divertor is one part of the DEMO fusion reactor. In DEMO, there are 54 cassettes and each cassette is supported by stainless steel structure, the weight of each cassette is around 7 tones. The divertor is devoted to extract the ash and heat generated during fusion reaction, minimize plasma contamination and protect the surrounding walls in the extreme environment where thermal and neurotic loads are existed.
Concept of remote handling robot is designed, and main components of the robot such as bearings, wheels, rails and cylinders are selected. FEM analysis on critical point especially contact area is carried out. Mechanism of the robot and cassette removal and installation sequences are studied.

Challenges here are space limitation and avoiding collision. The rail change system is one of the crucial part in this concept; four V shape wheels integrated with spherical roller bearing can rotate and roll along the toroidal rail. Clearance between blanket and the cassette is eliminated by the hydraulic jack installed on the top of the structure. The driven system of the structure movement along the toroidal rail is two telescopic cylinders, each cylinder controls one direction. Moreover, the merits and demerits are mentioned, in this concept, the structure is simple and have high stiffness, but the modification on the vacuum vessel should be taken into further consideration since the radiation may cause unexpected deformation on the rail.

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Experimental validation of Enhanced Heat Flux First Wall Panel Mechanical attachment system

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JSC NIKIET is a main supplier of ITER in-vessel components and its responsibility includes the manufacture of the FW beam, finger bodies, and the mechanical attachment system of the Enhanced Heat Flux (EHF) First Wall (FW) Panels in the framework of Procurement Arrangement 1.6.P1A.RF.01 dated 14.02.2014. The mechanical attachment system comprises the central bolt, threaded barrel and system washers. This FW mechanical attachment system allows the mounting of the FW onto the installed Shield Block (SB) while accommodating all external forces acting during plasma disruptions and normal operation. The design of EHF FW mechanical attachment system has been developed and optimized through a collaboration between the ITER Organization Central Team (IO-CT) and Russian specialists in order to provide the possibility to use it for all the FW panels. Full scale mockups of the FW attachment system components have been manufactured in JSC NIKIET in 2015-2016 and mechanical tests have been performed to demonstrate operation of these mock-ups under a 1 MN cyclic force. This paper summarizes the design description and experimental results of the FW attachment system.

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Reliability Assessment of remote maintenance strategy for CFETR Divertor

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China Fusion Engineering Testing Reactor (CFETR) will be built to test and verify the feasibility of engineering and technology in practice for the future fusion reactor. Long pulse and steady-state operation will be demonstrated with duty cycle time not less than 30~50%.During plasma operation, the in-vessel components of the fusion reactor will be activated and contaminated with tritium. Because of the beta and gamma activation of the component bulk and surface dust (beryllium, carbon, tungsten) special remote handling techniques will be required during machine maintenance periods. The divertor cassettes remote replacement is one of the key maintenance operations for the CFETR to meet the requirements of duty cycle time. RH maintenance strategies will have a significant impact on the layout of the machine and design of components. In this paper we will give an overview of the different CFETR divertor components remote maintenance strategies as well as to describe the maintenance process of the in vessel components. Preliminary assessment of the

divertor maintenance scheme is done in order to carry out RH maintenance tasks successfully and efficiently. Considering the number of feasible designs for the divertor maintenance, we concentrate remote handling concept assessments on the follow principles: As simple as possible; high-security; high reliability and availability.

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Post-test examination of a Li-Ta heat pipe exposed to H plasma in Magnum PSI

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The authors exposed a radiatively cooled, ~195-mm-long, lithium-filled tantalum heat pipe (HP) to a hydrogen plasma in DIFFER's linear plasma source Magnum PSI continuously for ~2 hours. We kept the overall heat load on the inclined HP constant, varied the tilt and peak heat flux to ~2.5 MWm2. The HP operated at ~1000-1100 C. Diagnostics included near infra-red thermography from two orthogonal ports. [1]

We stopped after an initial breach occurred near the beam axis intercept. ~0.06 g of lithium formed a 6-mm-diameter nodule and coating covered half of the area wetted by the beam. Several seconds later, a transverse crack opened; lithium flowed out quickly and wetted an area ~30 mm2. Fractography and post-test metallography showed differing breach mechanisms. The first site had prior material damage. The second breach occurred as the HP cooled. With creep and relaxation during the exposure, the site would have tensile loading during cooling. A tantalum disk annealed in a hydrogen furnace was available for comparative evidence of hydrogen embrittlement.

Sandia had purchased an existing tantalum HP from Aavid-Thermacore, Inc. The test showed prolonged operation, gave useful data and we judged it a success. However, tantalum would not be the right choice for a future PFC. The paper also discusses HP configurations and materials for future PFCs.

[1] G.F. Matthews, R. Nygren, T. Morgan, S.A. Silburn, "Demonstration of the potential for exchangeable PFCs based on radiatively cooled lithium heat pipes in Magnum PSI," this conference.

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Analysis of an actively-cooled coaxial cavity in a 170 GHz, 2 MW gyrotron using the multi-physics tool MUCCA

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Continuous Wave (CW) gyrotrons are the key elements for electron cyclotron resonance heating and current drive in present machines and future fusion reactors. In the frame of the EUROfusion activities, a 170 GHz, 2 MW short-pulse (ms) coaxial gyrotron existing at Karlsruhe Institute of Technology (KIT) is being upgraded for operation at longer pulses (100 ms - 1 s). In the coaxial gyrotron, the resonant cavity is made by a hollow cylindrical resonator and a coaxial inner insert, which permits increasing the gyrotron output power by enhancing the mode selectivity of the cavity. A forced flow of pressurized subcooled water, passing around the resonator as well as through the insert, is used to keep the cavity cooled.
The MUlti-physiCs tool for the integrated simulation of the CAvity (MUCCA) has been recently developed in collaboration between Politecnico di Torino and KIT. MUCCA is applied here to the analysis of the cavity of the upgraded 170 GHz, 2 MW coaxial gyrotron, to assess the evolution of its working point, starting from cold conditions, when different cooling configurations are considered for the resonator. An iterative procedure is adopted, where thermal-hydraulic and thermo-mechanical simulations are first performed on both resonator and insert with the commercial software STAR-CCM+, evaluating the mechanical deformation. The deformed cavity shape is then used as input in the electro-dynamic module EURIDICE to evaluate the heat load on the cavity wall, which becomes in turn the driver of the thermal-hydraulic analysis in the following time-step.
The main results of the MUCCA simulations are presented in terms of evolution of temperature, heat load, and deformation of the heated surface of the resonator and insert in the first seconds of operation. It is shown that the cavity behavior evolves towards a stable operation, with a maximum temperature strongly dependent on the cooling configuration.

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Integrated Power Exhaust Modelling for DEMO with Lithium Divertor

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Operation of a future demonstration fusion reactor (DEMO) requires the handling of a significant power flux that crosses the separatrix and enters the scrape-off layer. A considerable amount of energy has to be dissipated before the heat flux reaches divertor plates. The divertor may be exposed to high heat fluxes causing high temperature gradients and material fatigue. Such challenging conditions in the scrape-off layer demands developing solid state plate protection scenarios. In the liquid metal divertor, which is one of the considered solutions to the power exhaust problem, a liquid surface protects the solid plates against high heat loads, allowing for longer operation without the need of the divertor disassembly.
In this paper a DEMO reactor is considered with a liquid lithium divertor setup. The simulation is performed with the COREDIV code which self-consistently solves radial 1D energy and particle transport equations of plasma and impurities in the core region and 2D multifluid transport in the SOL. Influence of sputtering, prompt redeposition and evaporation of lithium is taken into account. An operational space of parameters is analyzed. Two regimes of operation are identified. The sputtering regime occurs when the divertor is tilted significantly with respect to the magnetic surface. Evaporation is low compared to the sputtering. In such a situation divertor power load is high and additional seeding is required in order to dissipate energy before it reaches the divertor plates. In the evaporation regime, evaporation is higher than sputtering. A high amount of lithium is released into the plasma diluting it and therefore reducing the power to the plates. Although in both regimes lithium dilutes the core plasma reducing fusion power, it is also fully stripped what results in low radiation and high power across the separatrix. Cooling of the plasma can then be achieved by seeding additional impurity.

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Design of the ITER EC upper launcher nuclear shielding

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ITER will be equipped with four EC (Electron Cyclotron) upper launchers of 8 MW microwave power each with the aim to counteract plasma instabilities during operation. These launcher antennas will be installed into four upper ports of the ITER vacuum vessel.
Beside their functional purpose the port plugs which are the structural system of the launchers have to provide as much shielding as possible in order to protect adjacent components from neutrons and photons and to minimize the shutdown dose rate in the port interspace and the port cell, being located further back in the ITER Tokamak building.
In addition to the basic design of the port plug cask several components are particularly shaped in order to achieve maximum shielding performance. Moreover three individual shield blocks will be installed at prominent positions inside the plug.
This paper presents the general design, the technical integration and mechanical stress analyses for all shielding components. Also the thermal-hydraulic properties and the integration of these water-cooled elements into the port plug´s cooling circuit are outlined.

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Exploratory risk analysis of ITER Cask & Plug Remote Handling System

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An exploratory risk analysis of ITER Cask & Plug Remote Handling System (CPRHS) has been performed under a system engineering approach considering the CPRHS in various operational states with the associated loads.
A Functional Breakdown Structure was developed from the 4 main functions fulfilled by the CPRHS: to dock, to handle, to transport and to confine. During Tokamak maintenance operations, various operating states and locations were defined for the CPRHS. In regard to the safety function, to confine, specific configurations were considered to capture all relevant loading cases.
A Failure Mode Effects & Criticality Analysis has been made by identifying potential failures of basic functions to be fulfilled by CPRHS during the maintenance of a Diagnostic Port Plug and quantifying them in terms of Criticality defined as the product of the failure Occurrence and Severity. The Severity rating scale was related to unavailability of the function due to both technical and safety issues.
Specific analyses of docking operations in the different cells and traveling operations in different rooms while taking into account the safety constraints were made leading to recommend actions for mitigating the failures having the highest criticality levels.
In order to perform a sensitivity analysis on maintenance duration by means of statistics approach, the failure modes of CPRHS basic functions were introduced in the schedule Primavera Risk Analysis software which uses a probabilistic Monte Carlo approach. The failure modes were considered as task dependent activities with a duration equal to Mean Time To Repair and an existence likelihood equal to the product ⊠ x DC where ⊠ is the failure rate and DC is the Duty Cycle of the failed basic function.

CPRHS availability and operations time were then addressed while considering the impact of the failure modes on maintenance operations time and the benefit of mitigation actions.

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First results from a new tritium capable ion implantation materials facility.

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A new tritium facility to study the interaction of tritium with fusion relevant materials, and its retention and release, has been produced. Tritium retention is a major issue for fusion power devices. The new facility allows implanting of a range of gases into samples, including tritium. This facility is currently used for the UKAEA led Tritium retention in Controlled and Evolving Microstructure (TriCEM) project, which includes modelling work investigating the interaction of hydrogen isotopes with different types of microstructural damage, validated by experiments. The experimental section includes sample production, irradiation, implantation with deuterium or tritium, and characterization. Self-ion bombardment with energies of several MeV is used to mimic the defects created by neutrons in fusion power plant and the created traps are then filled with D/T in the new facility. The samples are analysed primarily using Thermal Desorption Spectroscopy (TDS) and Secondary Ion Mass Spectroscopy (SIMS). Part of the characterization and analysis work is carried out at the new Materials Research Facility (MRF). There is a noted lack of tritium retention experiments that this new tritium facility will make up for. Accurate study of isotope effects, such as the isotopic exchange in damaged microstructure, has previously been difficult due to a background signal of light hydrogen. This new capability will allow virtually background free measurements using tritium and deuterium. The design and build of this facility is described. Commissioning results are presented along with the first results from materials with controlled damaged microstructure.

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Development of power combination system for high-power and long-pulse ICRF heating in LHD

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In the Large Helical Device (LHD), the development of high-power and long-pulse ICRF system is ongoing. Frequency was fixed at 38.47 MHz for the optimization of devices. At this frequency, plasma is heated with the minority ion heating of hydrogen and the second harmonic heating of deuterium. Field-Aligned-Impedance-Transforming (FAIT) antenna has the potential performance of high-power injection of more than 1.8 MW. In order to reduce the voltage in the transmission line, Ex-Vessel Impedance Transformer was also developed. However, the output power of the final power amplifier (FPA) is less than 1.2-1.3 MW for the stable oscillation in the LHD. In order to increase power into the FAIT antenna, a power combination system was developed. The target is injection power of 2 MW for 2-3 s and 1 MW for 10 min. By iterating the simulation of the electromagnetic field, optimized power combiner was designed. Then the combiner was fabricated and installed in the transmission system. The power combiner has two input ports and two output ports. Two FPAs are connected to the input ports. From one output port the combined power will be transmitted to

the FAIT antenna. Another output port is connected to a dummy load. Air ducts for the air cooling are attached for the long-pulse operation. It was confirmed by the measurement with the network analyzer that the power combiner has almost perfect isolation between input ports and there are no reflections from these ports when there is no reflection at the output ports. Control of power and phase of foreword waves into the input ports are important for the combination of the waves without power loss. Therefore, a real-time control system was developed and demonstrated. Step time of the control was less than 1 ms and power loss into the dummy load was successfully cancelled.

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EM analyses on the 55.NE.V0 loom system and attached components

Co-author: Claudio Bertolini [1]

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The ITER vacuum vessel (VV) contains transducers for vessel and blanket instrumentation and for the measurement of plasma performance. The electrical and optical signals to and from these transducers are managed through the electrical services infrastructure project 55.NE.V0. This system is responsible for transmitting electrical signals from the VV inner skin to the outside of the vacuum where they interface to conventional signal transmission and data acquisition systems. Around 1400 sensors/junction boxes/connectors and more than 1700 MI cables are distributed throughout the vacuum vessel inner surface.

This paper concerns EM activities performed as part of the full set of engineering analyses necessary to assess the design configuration feasibility.

Transient electromagnetic analyses were performed for 4 representative looms systems, placed at the inboard and outboard region, and exiting the VV from lower and upper ports. Highly populated looms in the divertor region (named 55.NE.D0) were analysed for the most demanding downward plasma disruptions.

All the cables were modeled using dedicated EM "wire elements", an element type developed by LTCalcoli, suitable to model eddy currents in wire-shaped structures.

The calculated distributed EM loads along the cables and in attachment components allowed the most loaded cable branches, and the worst position and disruption for connecting clamps and junction boxes, to be identified.

Detailed 3D EM models were developed for the most loaded junction boxes and clamps and integrated in the ITER Global Model.

The detailed analyses allowed the Lorentz forces and resultant torques on the different sub-structures, and the force distribution element by element, to be evaluated, ready for subsequent structural assessments.

Since the clamp bosses are welded onto the VV which is a confinement boundary and ESPN equipment, the analyses conducted contribute to demonstrating the absence of impact on the VV. This work is considered to be a Protection Important Activity (PIA).

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A novel code for the simulation of plasma equilibrium and evolution

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The simulation of the plasma equilibrium and its evolution is important for the study of plasma physics and for the correct design of fusion devices. For this purpose, a novel code, based on the solution of the Grad-Shafranov equation, has been fully implemented in ANSYS. It exploits the finite element method using the magnetic potential vector formulation. In this approach plasma pressure and current density profiles are described by means of two main parameters: the internal plasma inductance li and the Poloidal Betap. Several ITER equilibria of limiter and diverted plasma configurations have been reproduced and benchmarked with validated codes such as MAXFEA and DINA. Being fully implemented in ANSYS this code will allow the coupling of the 2D axisymmetric plasma equilibrium equations, describing the plasma behavior, with the full 3D eddy currents equations, describing the surrounding three-dimensional structures.

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Modelling of MAST-U neutral beam re-ionisation and the impact on the beamline ducts and in-vessel components

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Neutral beam injection is one of the primary auxiliary heating systems for tokamak plasmas. Once the neutral beam leaves the neutraliser collisions with background neutral particles in the beamline and tokamak vessel re-ionises part of the neutral beam. These particles can be deflected by the tokamak magnetic field, potentially damaging unshielded components.
The first stage of the Mega Amp Spherical Tokamak Upgrade (MAST-U) has two Positive Ion Neutral Injectors (PINIs), one injecting power to be deposited close to the magnetic axis of the plasma (on-axis) and one injecting power that is deposited further out in the plasma (off-axis). Each injector has been designed to run for up to 5 seconds, with 2 seconds of neutral beam required for MAST-U core scope. The increase in neutral beam pulse length and the increase in complexity of the in-vessel hardware compared to MAST increases the risk posed by re-ionisation to the beamline ducts and in-vessel components.
This paper describes modelling of the neutral beam re-ionisation along the beamlines and into the MAST-U vessel, within a field envelope applicable to the operational space of the machine. Monte-Carlo simulations using the MAGNET code give the re-ionised power loading on both beamline ducts, which was used to position thermocouples on the duct liners. Thermal and structural analysis of the duct liners has been carried out using ANSYS.
The potential for significant amounts of re-ionised power entering the MAST-U vessel has resulted in the installation of graphite tile plating at the end of the ducts, compatible with co-injection. The power loading on these plates and other in-vessel components as modelled by MAGNET has been verified using the LOCUST code. Analysis has confirmed that the graphite tiles are compatible with core scope operation.

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Progress in the pre-conceptual CAD engineering of European DEMO divertor cassette

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This paper presents the recent progress in the pre-conceptual design activities for the DEMO divertor Cassette Body, performed in the framework of the work package “Divertor” of the EUROfusion

Power Plant Physics & Technology (PPPT) program. According to Systems Engineering Principles, the divertor CAD model is reviewed, considering the updates in the DEMO configuration model presented by the Programme Management Unit (PMU) in 2017. The design parameters affected by these changes and their impact on the divertor design and on the interfaced systems are analysed. Then, the paper focuses on the integration on the new cassette geometry of the divertor sub-systems. This includes the design of a "shielding liner" for cassette body and Vacuum Vessel protection, as well as the development of the cassette body-to-Vacuum Vessel fixation system. The design activities related to these main sub-systems are discussed in detail, in terms of CAD model and thermo-mechanical calculations.

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Experimental evaluation of wall shear stress in double contraction nozzle for structural soundness evaluation for liquid Li target of intense fusion neutron source

Co-author: Hiroo Kondo [1]

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This paper reports on an experimental evaluation of wall shear stress in a double contraction nozzle to produce a liquid lithium (Li) target being intended for use as a beam target for the intense fusion neutron sources such as the International Fusion Materials Irradiation Facility (IFMIF), the Advanced Fusion Neutron Source (A-FNS), and the DEMO Oriented Neutron Source (DONES). The current design of the neutron sources requires that the thickness of the liquid Li target be maintained to within ± 1 mm while operating under normal conditions (target speed: 15 m/s, inlet Li temperature: 250 ℃, and vacuum pressure: 10−3 Pa). In the IFMIF/EVEDA project, we designed and constructed the IFMIF/EVEDA Li Test Loop (ELTL), and we performed experiments to validate the stability of the Li target. After the end of this project, the ELTL has completed its disassembly work in Feb. 2017 and structural soundness evaluation of the configuration components of the ELTL is being prepared.
Prior to the structural soundness evaluation, we have investigated the flow characteristics in the nozzle which is a key component producing the stable Li target. The wall shear stress is an essential physical parameter to understand erosion-corrosion by a high speed liquid Li flow, and therefore we evaluated by using an acrylic mock-up of the target assembly experimentally. The working fluid was water whose kinematic viscosity coefficient is similar to that of liquid Li. The velocity distribution in the nozzle was measured by laser-doppler velocimetry and momentum thickness along the nozzle wall was calculated by Buri's equation. Based on the calculated momentum thickness, we estimated the variation of the wall shear stress along the nozzle wall and showed an importance indicator for erosion-corrosion of the nozzle for the future structural soundness evaluation.

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Strategies toward the Realization of the Helical Fusion Reactor FFHR-c1

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Design studies on the helical fusion reactor FFHR-c1 has been progressed. The main goal of the FFHR-c1 is to demonstrate one-year steady-state sustainment of the fusion plasma with self-produced electricity and tritium. The major radius of the plasma, R, is ~10 m and the magnetic field strength at the plasma center, B, is ~8 T. High-temperature superconductor (HTS) magnet coils are adopted in the design. During the one-year operation, ~370 MW of fusion power is sustained with ~25 MW

of auxiliary heating power supplied by the electron cyclotron heating (ECH) of ~220 GHz. Then, the fusion gain, Q, is ~15. A liquid metal divertor system named the REVOLVER-D is adopted for the reasons of easy maintenance and a high permissible heat load. In this system, ten sets of molten tin showers are discretely inserted to the inboard side of the ergodic layer surrounding the main plasma. A cartridge-type molten salt (or liquid metal) blanket named the CARDISTRY-B is also adopted for easy maintenance. For the realization of the FFHR-c1, it is necessary to accumulate experiences on the new technologies of the HTS magnet coils, the REVOLVER-D, and the CARDISTRY-B. Including these three, we have defined 22 important issues that should be resolved before building the FFHR-c1. The strategy to efficiently address the 22 issues has been also discussed. As a result, we propose a step-by-step approach by developing FFHR-01 (R ~ 0.4 m and B ~ 3 T) for basic studies on HTS coils and the REVOLVER-D, FFHR-a1 (R ~ 2.5 m and B ~ 4 T) for demonstration of one-year operation under a non-nuclear condition, and FFHR-b1 (same R and B as FFHR-a1) for DT operation, before FFHR-c1. The FFHR-b1 plays a role of volumetric neutron source (VNS) and the FFHR-a1 corresponds to the cold test of the VNS.

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Measurement of neutron fluence in the High-Flux Test Module of the Early Neutron Source by an activation foils method

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The neutron fluence is an important normalization parameter for the material specimens to be irradiated in the Early Neutron Source (ENS). The activation foil method appears suitable for this purpose considering cost, low technical requirements and invasivness.
Small packages of thin activation foils can be placed in several locations: on the outer surface of the HFTM, on the outside of specimen capsules inside the HFTM or inside the specimen capsules. The latter would provide measurements very close to the specimens while the other two options require more corrections to determine the neutron fluence in the place of the specimen. Each location has a different access time after completion of the irradiation cycle. If the activation foils are mounted on the outer surface of the HFTM the estimated earliest access to them for measurement of the induced gamma activity would be approximately one week after shut-down. An activation foils package on the surface of a specimen capsule would be accessible about three weeks after shut-down while an activation foil package inside the specimen capsule becomes available two to three months after shut-down.
In this work we will present a set of activation foils which is suitable for application in the HFTM. The set consists of iron, cobalt, nickel, yttrium, and gold. The selected dosimetry reactions lead to radioisotopes with half-lives of several months up to a few years so that they preserve neutron flux information over the full irradiation time, and they cover the entire ENS neutron energy range. Tests of the measurement method were performed with the cyclotron neutron source at NPI Řež which provides a fast neutron spectrum similar to ENS. We will present the analysis of these tests together with a review of state-of-the-art evaluated cross sections of the dosimetry reactions of interest.

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Assessment of controllers and scenario control performance for ITER first plasma

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Assessment of Controllers and Scenario Control Performance for ITER First Plasma

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The ITER PCS is a key component of ITER operation, with performance requirements much more stringent than existing devices. We report on assessment of control algorithms and control scenarios comprising the prototype ITER PCS design, which is the starting point for development of the final design for first plasma operation. The scenarios assessed include commissioning of magnetics and gas systems using the PCS and the first plasma scenario, which includes neutral gas prefill, plasma breakdown/burnthrough, and initial evolution of equilibrium and plasma density. Systems involved in first plasma control include ECH, PF and CS coils and power supplies, gas valves, and magnetic, neutral pressure, and electron density diagnostics.

Assessment involves simulation of an ITER PCS model connected in feedback with an ITER plant model, both executing in the Plasma Control System Simulation Platform (PCSSP). PCSSP is presently undergoing upgrades as part of PCS development to provide support for algorithm development, PCS architecture evaluation, and control performance assessment. In particular, PCSSP provides general methods for extensive testing of performance in the face of multiple adverse events, such as plasma instabilities growth, disruptions, or plant system faults. Definition and application of performance metrics to control simulation results will be discussed.

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Conceptual design of a Neutral Beam Heating system for DTT

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The main purpose of the Divertor Tokamak Test (DTT) is to study solutions to mitigate the issue of power exhaust in conditions relevant for ITER and DEMO. The key feature of such a study is to equip the machine with a significant amount of auxiliary heating power (45 MW) in order to test different divertor solutions. According to the Italian project, the experiment is foreseen to operate with the following main parameters: BT = 6 T, IP = 5.5 MA, R0 = 2.08 m, a = 0.65 m and a pulse duration of 90-100 s. It shall be able to study different divertor magnetic configurations and reach a reactor relevant power flow to the divertor. The proposed mix of heating power foreseen to achieve the target value of 45 MW delivered to the plasma will be provided by Electron Cyclotron Resonant Heating (ECRH), Ion Cyclotron Resonant Heating (ICRH) and Negative-ion-based Neutral Beam Heating (NNBH).
In this framework, the conceptual design of a NNBH system for DTT is here presented, with a

particular focus on the technical solutions adopted to fulfil the requirements and maximize the performances. The proposed system features two beamlines providing deuterium negative ions (D-) with an energy not smaller than 300 keV and an injected power of 5-8 MW each.
The design of the main components of the injectors is described in detail, explaining the motivations behind the main design choices. A comprehensive set of simulations was carried out using several physics and engineering codes to drive the development of the design. These simulations mainly regard the efficiency of the main processes, the optics of the beam, the physics reactions along the beamline (stripping, charge-exchange and ionization), the thermo-mechanical behaviour of the acceleration grids and the coupling between the beam and the plasma in the tokamak chamber.

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Effect of plasma screening on pumping efficiency in the DEMO divertor

The screening of a neutral gas by plasma from the top of the private flux region (PFR) of the latest DEMO divertor configuration without the dome structure is analysed. The effect of the neutral gas compression in the PFR is assessed by using the direct simulation Monte Carlo method (DSMC) and the ambipolar approximation for the simulation of neutral molecule dissociation and ionization as a result of collisions with plasma electrons.
Preliminary results without the divertor dome have been reported previously [1], in which the atomic and molecular processes as well as the interaction with plasma were neglected. It was shown that a strong neutral reflux from the private flux region towards the x-point occurs. The aim of the present work is to investigate the impact of neutral interaction with electrons on particle reflux. The plasma fan could impede the outflow of neutrals to the bulk plasma due to the sharp electron density and temperature drop towards the PFR, resulting in an effective neutral plugging.
The numerical analysis includes the calculation of neutral density, temperature and pressure in the divertor plenum and the overall conductance of the sub-divertor structure, which consequently affects the pumping efficiency. It is shown that plasma plugging is more efficient in the case of plasma attachment. Analysis of dome structure requirements for achievement of the effective pumping is extended by including the plasma effect as a plugging of neutrals in the PFR. The DEMO divertor cassette structure can be further optimized to ensure more efficient pumping and achievement of detachment conditions.

[1] Chr. Day, S. Varoutis, Yu. Igitkhanov, IEEE Trans. Plasma Science 44 (2016) 1636-1641.

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Automation of upgraded NBI cooling water system

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NBI Cooling Water System is used during beam operation for the removal of received heat load from vessel sub-components. During the process of shifting the NBI Vacuum Vessel to SST-1, it was needed to shift the cooling water plant. Shifting of Cooling Water Plant leads to the dis-mantling of the actual plant and designing, development and installation of a new plant. The cooling water plant is designed for a capacity of 2895 lpm (max.) with a pressure upto 11 bar (for grids loop).

Main goal of realizing this system is the full Automation; which is achieved by using Step 7 programming for Programmable Logic Controller (PLC), Supervisory Control and Data Acquisition System (SCADA) GUI for the Control logic and data logging & TIA Portal for the PLC Touch Panel (HMI)

along with necessary hardware. The paper describes the hardware tasks and the software tasks accomplished for the automation of the cooling water system.

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Factory acceptance test results of ITER EU ECPS

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The power supply set for the EU EC Heating system (ECPS) of ITER provides up to 6 MVA electrical power to two 170GHz/1MW Gyrotrons. The required electrical power for the gyrotrons is both very high and has to comply also with highest quality requirements. These performance indicators were proven with full voltage modulation at rates up to 5kHz.

Ampegon's newly developed power supply topology is optimized to cope with these stringent requirements. Two different topologies are combined. On one hand, the PSM topology for the Main High Voltage Power Supply (MHVPS) and on the other hand the enhanced PSM topology for the two Body Power Supplies (BPS). The enhanced PSM topology demonstrates an improved accuracy and very low ripple values. Furthermore, this topology is designed to supply capacitive loads. Besides the demanding dynamic requirements and ripple performance, the power supplies must protect the Gyrotron in case of an arc. Therefore, the energy into the arc is also an important figure for the qualification of the power supply.

To test the power supply set (one MHVPS and two BPS) in its various operating modes, four different dummy loads have been designed and are part of the Ampegon scope of supply. The complete set, including the relevant control interfaces, has been installed and tested in Ampegon's test laboratory. The test setup and the factory acceptance test results for the ECPS Heating System are presented.

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Thermal diffusivity of ceramic breeder beds

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In the solid Breeder Blanket (BB) concepts both tritium release and heat recovery depend on the thermal performances of the breeding zone. Within the R&D activities of the Helium Cooled Pebble Bed (HCPB) breeding blanket, the knowledge of the thermal diffusivity of the breeder beds is of fundamental importance to model the transient heat transfer during the power pulses of the fusion machine. The aim of the present study is to investigate the thermal diffusivity of the breeder beds at

BB relevant conditions; to this end the line heat source probe method was employed. The method uses a linear heater (probe) embedded in the material to be investigated. The thermal diffusivity of the material is analytically derived by measuring the temperature rise as a function of time at a point close to the probe. By knowing the thermal conductivity of the material, the specific heat capacity can be derived in addition.
An experimental facility was conceived for the investigation of the thermal diffusivity of granular beds at breeder blanket relevant temperatures, mechanical state, purge gas type and pressure. Based on the geometrical restriction of the experimental facility, the experimental parameters were tailored with a series of FEM simulations to reduce the error of the measurements.

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Design concept and thermal-structural analysis of a high power reflective mm-wave optical mirror (M2) for the ITER ECH Upper Launcher

Co-authors: Philip Santos Silva [1]; Rene Chavan [1]; Timothy Goodman [1]; Avelino Mas Sánchez [1]; Matteo Vignoni [1]

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Each of the 4 ITER Electron Cyclotron Heating Upper Launcher (ECHUL) features 8 transmission lines (TLs) used to inject 170 GHz microwave power into the plasma at a level of up to 1.31 MW (at the TL diamond window) per line. The millimetre waves are guided through a quasi-optical section consisting of three fixed mirror sets (M1, M2 and M3) and one front steering mirror set (M4).
The M2 mirror set is composed of an upper and lower part, each reflecting 4 nearly-Gaussian beams coming from the M1 to the M3 mirror, which focuses them towards the M4 steering mirror that will aim at the correct location in the plasma for suppression of the q=3/2 and q=2/1 NTMs.

Mm-wave power is converted into heat by ohmic dissipation, reaching a peak power density of approximately 4 MW/m2 on each of the 4 beam centre spots of the M2 mirror, resulting in a total of 19.4 kW absorbed power.

EPFL-SPC has developed a novel water cooled mirror design concept which is able to dissipate such high heat loads (with up to 60000 thermal cycles) and also resist the applied external loads and dynamic displacements arising from plasma disruptions and seismic events, while complying with: material, manufacturing and space restrictions.

This study describes the main design features of the M2 upper mirror, and its design conformity in accordance to the ASME design code and also its conformity to the Essential Safety Requirements (ESR) for Nuclear In-vessel components.

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Material Irradiation Tests in the ITER Divertor Relevant Settings

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Various types of multilayer laser mirrors and piezoelements underwent radiation tests to assess the influence of neutron and gamma-ray fluxes similar to those expected in diagnostic ports of ITER divertor. The optical and thermal performance of laser mirrors and the piezoelectric coefficient of the piezo-elements were under investigation. The test was performed in the RIAR irradiation facility RBT-6 using ITER relevant neutron and gamma fields with a neutron flux of ~10^12 n/cm^2/s and a fluence of ~10^19 n/cm^2 (E > 0.1 MeV); heating did not exceed 200 C. Special racks for arranging the test samples of mirrors and piezo-elements in the reactor were designed with neutron shields for adjusting neutron and gamma-ray spectra; also, helium atmosphere for the in-situ cooling was used. The neutron spectra, estimated using MCNP and tested experimentally, are provided and compared with those expected in ITER under baseline inductive burning plasma conditions (Q = 10). The tested laser mirrors (40 samples of 4 types) have a metallic sublayer (Al or Ag) and a multilayer dielectric coating (ZrO2 / SiO2 or Sc2O3 / SiO2). The mirrors are designed for broadband reflection and high laser power breakdown threshold at the wavelengths 956nm, 1047nm and 1064nm (high-power lasers of divertor Thomson scattering), as required for the combined diagnostics 55.EA (laser-induced fluorescence) and 55.C4 (Thomson scattering). The mirror breakdown threshold and heating resistance up to 200-250 C were tested before irradiation; further tests are planned. The piezoelectric coefficient of the two piezo-ceramic types chosen for stick-slip and resonance piezo actuator types was measured before and after irradiation. Experimental approaches and facilities used for irradiation, and pre- and post-irradiation measurements are described in detail. The first results and outlook of further experiments are also presented.

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Tokamak T-15MD - two years before the physical start–up

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At the present time, in the NRC Kurchatov Institute within the Federal Target Program "Nuclear energy-technologies of new generation for period 2010 - 2015 and to the prospect until 2020" the tokamak T-15MD with supporting facilities is being built. The preassembly of the tokamak T-15MD magnet system together with vacuum chamber was completed at a plant in Bryansk. All elements of the magnet system and vacuum chamber will be delivered to the NRC "Kurchatov Institute" in Moscow for the tokamak T-15MD assembly. It is expected that the T-15MD assembly will be completed by the end of 2018. The reconstruction of sub-station 110/10/0,4 kV for own needs was completed in 2017 and the reconstruction of the main sub-station 110/10/1 kV, 300 MW is completed in 2018 year. Twenty- two of the new transformers 10/1 kV and the same new thyristor convertors will be installed during the 2018-2019. One gyrotron with output power 1 MW for pre-ionization should be installed in 2019. Tokamak T-15MD connection to water and electrical communication and also the adjustment of control system will be completed in the middle of 2020. Physical start-up T-15MD is scheduled for December 2020 year.

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Path planning and space occupation for remote maintenance operations of transportation in DEMO

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The ex-vessel Remote Maintenance Systems in the DEMOnstration Power Station (DEMO) are responsible for the replacement and transportation of the plasma facing components. The ex-vessel operations of transportation are performed by overhead systems or ground vehicles. The time duration of the transportation operations has to be taken into account for the reactor shutdown. The space required to perform these operations has also an impact in the economics of the power plant.

A total of 87 trajectories of transportation were evaluated, with a total length of approximately 3 km. The total occupancy volume is, comparatively, between 21 and 45 Olympic swimming pools, depending mainly to the type of transportation adopted in the upper level of the reactor building. Taking into account the recovery and rescue operations in case of failure, the volume may increase up to, between 43 and 64 Olympic swimming pools. The estimation of the total time duration of all expected transportation missions in the reactor building are between 166 hours (7 days) and 388 hours (16 days). This time estimation does not include docking, accelerations or other operations that are not transportation. The travel speed is assumed constant with a maximum value of 20cm/s (the same value assumed for Cask and Plug Remote handling System in the International Thermonuclear Experimental Reactor - ITER).

The results achieved in this preliminary assessment will help the design process to optimize the time duration of the reactor shutdown and the layout of the DEMO power plant.

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Characterization of the SPIDER Cs oven prototype in the CAesium Test Stand for the ITER HNB negative ion sources

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The ITER Heating Neutral Beam (HNB) injector is required to deliver 16.7 MW power into the plasma from a neutralised beam of H-/D- negative ions, produced by an RF source and accelerated up to 1 MeV. To enhance the H-/D- production, the surface of the acceleration system grid facing the source (the plasma grid) will be coated with Cs to reduce its work function. Cs will be routinely evaporated in the source by means of specific ovens embedded in the source. Controlling and monitoring the evaporation rate of Cs inside the source will be fundamental to get the desired performances on the ITER HNB.
In order to properly design the source of the ITER HNB and to identify the best operation practices for it, the prototype RF negative ion source SPIDER has been developed and built in the Neutral Beam Test Facility at Consorzio RFX. In SPIDER, liquid Cs based ovens will be used to inject Cs vapours inside the source. The CAesium Test Stand (CATS) has been specifically designed and set up for testing, commissioning, and characterizing Cs ovens in vacuum, but also to study the Cs evaporation and deposition onto surfaces. A SPIDER Cs oven prototype has been manufactured and tested in CATS in order to characterize its thermal behavior, by means of thermocouples and thermal camera, and its Cs flux, by means of Surface Ionization Detector and Laser Absorption Spectroscopy.
The paper will present the CATS set up with a description of the layout and main features. The paper will also show the experimental results on the characterization of the Cs oven prototype for SPIDER.

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Testing of ceramic membranes for PEG separation and preliminary design of a membrane cascade

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The Plasma Exhausts Gases (PEGs) proposed to reduce the power load over the plasma facing components are separated by the Plasma Exhaust Processing System of DEMO.
Two kinds of ceramic porous membranes (with top layer of pore size 0.2 ⊠m and 3-4 nm, respectively) used commercially for the filtration of liquids have been tested in order to verify their application for the PEG separation. The experiments have been carried out at room temperature with feed pressure in the range 100-180 kPa and permeate pressure of 100 kPa by testing Ar, N2, He and H2. The results of the experiments have been exploited to validate a gas mass transfer model taking into account the permeation mechanisms of Knudsen and Poiseuille and, then, the model has been used to assess the permeance of the molecule DT.
According to the processing requirements for the PEG separation of DEMO and based on the permeance and selectivity values of the commercial membranes calculated by the model, a membrane system consisting of ceramic membranes (top layer of pore size 0.2 ⊠m) followed by a Pd-Ag permeator has been assessed. The calculation of a counter-current recycle cascade of ceramic membranes porous membranes has been conducted via the simplified Underwood-Fenske method under the hypothesis to use Ar as PEG. By vacuum pumping the permeate of the ceramic membrane cascade at 10 mbar the maximum Ar concentration in the retentate is 99.95%, while the higher Ar concentration of 99.995% can be achieved with a vacuum level in the permeate of 1 mbar. In both the cases, the Pd-permeator downstream the ceramic membrane cascade recovers about the 99% of the DT fed in form of ultrapure gas.

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Assembly and final dimensional inspection at factory of the JT60-SA Cryostat Vessel Body Cylindrical Section

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The JT-60SA project, a superconducting tokamak developed under the Satellite Tokamak Programme of the Broader Approach Agreement between EU and Japan and of the Japan Fusion National Programme, is progressing on schedule towards the first plasma in 2020. Within the European contribution to JT-60SA, Spain is responsible for providing JT-60SA cryostat.

The JT-60SA cryostat is a stainless steel vacuum vessel (14m diameter, 16m height) which encloses the tokamak providing the vacuum environment (10-3 Pa). It must withstand the external atmospheric pressure during normal operation, and internal overpressure in case of an accident (0.12 MPa absolute). The cryostat design is subdivided, for functional purposes, in two large assemblies: the Cryostat Vessel Body Cylindrical Section (CVBCS) and the Cryostat Base (CB). For transport and assembly reasons the cryostat is made up of 20 main parts: 7 making up the CB and 13 making up the CVBCS (including the top lid). All of the joints between them rely on bolted flanges together with light seal welds, non-structural fillet welds performed from inside and/or outside of the cryostat. The single wall is externally reinforced with ribs to support the weight of all the ports and port plugs and

also to withstand the vacuum pressure. The material SS 304 (Co<0.05 wt%) with a permeability (μrel) below 1.1. The CVBCS is made of a single wall stainless steel shell with a thickness of 34mm.The CB was manufactured and assembled in-situ in 2013, while the CVBC was manufactured by a Spanish company (ASTURFEITO S.A) and delivered to Japan in November 2017.

This paper summarizes the assembly and final dimensional inspection at factory of the JT-60SA CVBCS.

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Mechanical properties of dissimilar TIG welding for ARAA and SS316L

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Korea has designed a helium cooled ceramic reflector (HCCR) breeding blanket for developing the Korean DEMO and fusion reactor, including the development of the reduced activation alloy, ARRA (Advanced Reduced Activation Alloy). From the lesson of the developing procedure of the HCCR test blanket module (TBM) for ITER, it is known that the various fabrication methods, such as electron beam welding, TIG welding, and HIP (Hot Isostatic Pressing) bonding, should be developed for the fabrication of some complex structures. In the current research, the developed dissimilar TIG welding using ARAA and SS316L was introduced. The tensile, impact, bend tests were determined after post-weld heat treatment (PWHT) and hardness, microstructure characteristics were determined before and after to evaluate the welded specimen under the determined welding conditions. The average hardness values before were 338 HV in the HAZ and 199 HV in the WM, and after PWHT, the values decreases to 243 HV in the HAZ and 221 HV in the WM. Additional tests are underway for the ductile brittle transition temperatures (DBTT) on the heat affected zone (HAZ) and on the weld metal (WM). Tensile strength has decreased as the increase of temperature. The yield strength (YS) and ultimate tensile strength (UTS) were 286 MPa and 572 MPa, respectively

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Hydrogen isotope permeation experiment - design and first results

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Following the decommissioning of JET, and other future fusion reactors, there will be large amounts of tritiated waste requiring disposal. An appropriate containment strategy is required for storage of this waste. Studies have so far demonstrated that stainless steel appears to be the most promising containment material, but little is known about the permeation of hydrogen isotopes through stainless steel in waste relevant conditions. It is essential that this process is understood prior to the decommissioning of JET and start-up of ITER and DEMO.
An experiment has been designed to replicate waste storage relevant conditions and allow for the study of hydrogen isotope permeation under these conditions. The effect of temperature, humidity and surface uniformity of the steel are being studied for protium, deuterium and tritium.
This unique experimental rig has been built at UKAEA, to support JET decommissioning research funded by the UK Nuclear Decommissioning Authority, and is now carrying out testing with deuterium. The first results have been extremely surprising, with deuterium permeation significantly lower than expected for unbaked stainless-steel samples. This phenomenon is being further investigated to determine the cause, but is currently thought to be due to the protium content of the steel preventing diffusion of other hydrogen isotopes. This theory is supported by further experimental

results indicating that baking the samples at 400℃ to remove protium prior to commencing experimental work increases the permeation dramatically, to be in line with computational models. X-ray photoelectron spectroscopy and thermal desorption spectroscopy are being conducted to further understand the nature of the samples and rule out surface contamination or other unexpected effects. The outcomes of this work are entirely unexpected; no published literature appears to have investigated this phenomenon. The results are expected to have wide-ranging consequences for the storage and containment of tritium and tritiated waste worldwide.

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The remote handling systems of ifmif-dones

Co-authors: Gioacchino Miccichè [1]; Mark Ascott [2]; Ante Bakic [3]; Davide Bernardi [1]; Jose Brenosa [4]; Sofia Coloma [5]; Oliver Crofts [2]; Giuseppe Di Gironimo [6]; Manuel Ferre [5]; Georg Fischer [7]; Angel Ibarra [8]; Attila Karap [9]; Istvan Kiss [9]; Christoph Kunert [7]; Luciano Lorenzelli [10]; George Mitchell [2]; Martin Mittwollen [7]; Paolo Pagani [7]; Stefano Papa [6]; Garbor Porempovics [9]; Tonci Tadic [11]; Toth Matyas [12]

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The DEMO Oriented Neutrons Source (DONES) consists of complex systems and massive components that need to be on site assembled and maintained. 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 DONES' systems and components, located mainly in the Test Systems (TS), in the Lithium Systems (LS) and in the Accelerator Systems (AS), is classified as a remote handling (RH) Class 1st activity and as such is considered a crucial and essential activity whose success will strictly depend on the DONES RH capability. According to this, a Remote Handling Systems (RHS) for DONES, which comprises the whole set of Remote Handling Equipment (RHE) and tooling for the execution of maintenance tasks, has been designed. A wide range of technologies is involved: special cranes, manipulator arms, lift interface frames, special cameras, control systems and virtual reality.
In this paper an overview on the design of the main robotic systems and tooling of the RHS of DONES, including design requirements, functions and maintenance tasks to be performed, is given.

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

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Purification of Pb-16Li breeder from corrosion products

Co-author: Lukáš Košek [1]

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This contribution provides summary of two purification experiments of liquid metal breeder Pb-16Li by a cold trap. The behavior of artificially added impurities were studied in non-isothermal ferritic loop Meliloo v1. During these experiments a Mn concentrations followed the solubility curve as published by Barker. More advanced trap design was tested in austenitic loop Meliloo v2. This trap was analyzed for deposits distribution over the internal surfaces using SEM. Large Fe-Cr-Mn-Ni particles and Ni-Mn, Ni-Mn-Sn intermetallic phases were found on the cooled wall, while deposits of Bi oxides were found on the non-cooled internals. The SEM images of typical deposits are presented in the contribution along with the location where they were found.

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Effects of process variables on microstructure and tensile properties of friction stir welded ARAA

Co-author: Young-Bum Chun [1]

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Plasma facing component such as breeding blanket and divertor in fusion reactors are supposed to be assembled by welding and joining of parts made of reduced-activation ferritic-martensitic (RAFM) steel, and accordingly the structural integrity is significantly affected by the properties of the joint. Conventional fusion welding results in a wide heat-affected zone (HAZ) where a well-known type IV cracking occurs in RAFM steels under creep condition. In the present work, an attempt was made to butt weld the ARAA (advanced reduced-activation alloy; the Korean RAFM steel) sheets by the friction stir welding (FSW) technique, for which we investigated how the welding condition and post-weld heat treatment (PWHT) affect the microstructure and tensile properties of the joint. Microstructure of the as-welded ARAA was found to consist of a martensitically transformed zone (MTZ), a mixed zone (MZ) of fresh martensite and heat-affected tempered-martensite structures, HAZ, and base material in order outward from the welded interface. The hardness measurements showed a significant hardening in MTZ, an abrupt decrease of hardness towards HAZ in MZ, and the lowest hardness in HAZ. It is noteworthy that increasing the rotational speed and applied load, or decreasing the traveling speed increased the width of MTZ and reduced that of HAZ, which in turn led to increases of the tensile yield strength and ductility of the as-welded ARAA. PWHT conducted at 700 C resulted in transformation of martensite to tempered-martensite in MTZ and MZ while no significant change occurred in other zones. Such changes of microstructure reduced the gradient of hardness between MTZ and HAZ and the degree of strain-localization in HAZ. It is proposed that FSW technique can be utilized for butt welding of thin RAFM steel parts.

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Convective Baking Test of the ITER Lower Port for Factory Acceptance

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The ITER lower port is designed to support divertor remote handling and vacum pumping. To meet the purpose, it will be assembling to each main vessel on the vacuum vessel manufacturing site. Before delivering to the sector shop, a series of fuctional and mechanical test, which is so-called

factory acceptance test (FAT) should be performed by the manufacturer. The ITER FAT should be complying with the RCC-MR 2007 code and French regulations of neclear pressure equipment (ESPN) to assure quality of nuclear pressure equipment. The precedure of lower port FAT has been set to that visual test, pressure test, baking, vacuum leak test, and final dimensional inspection. Especially, the baking is critical cleaning method to satisfy required vacuum condition. The baking condition is challengeable to satify both the given ramp-up/down condition, which is 5 ℃/hr, and the temperature difference of the object within 40 ℃, simultaneously. The air heating and circulating furnace has been specially designed to apply convective heating and cooling method. This is because the lower port is large complex double wall box structures, convevtive heating and cooling is relatively proper method to satify the given condition with time. In this study, using the mock-up of the lower port, convective baking test is perfomred with maximum holding temperature is set to 220 ℃ during 12 hours. As a result, the maximmum temperature difference of the mock-up is apperaed 15 ℃, which is appeared at the ramp-down condition. In addition, applying the measured result, thransient thermal-structural analysis is performed. The maximum stress of 85.5 MPa is occured, and this is reasonable stress intensity compare to the allowable stress 187.5 MPa. It is a proven that the convective baking method is quite feasible as the FAT baking for the ITER lower port.

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Design and preliminary testing of a sieve tray column for PbLi purification

Co-author: Michal Kordac [1]

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Future designs of fusion devices are going to make use of a tritium production systems, several of which are considered using PbLi alloy as a breeder. Apart from tritium, other volatile and non-volatile species are being formed, either as products of the neutron irradiation or as corrosion products. The volatile impurities must be eliminated based on safety concerns (e.g. polonium) or blanket system malfunctions concerns (e.g. helium). For these, a gas-liquid contactor appears to be a suitable unit. A promising arrangement of the gas-liquid contactor is a column using a sieve tray as liquid PbLi distributor allowing certain free falling height for the desired purification to take place. This contribution presents details of the design and initial testing of such unit with the aim to remove highly volatile compounds by desorption into a stream of inert gas at ambient pressure.

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A two colors interferometer for PROTO-SPHERA experiment

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PROTO-SPHERA (Spherical Plasma for Helicity Relaxation Assessment) is a new concept of torus that aims to produce a Spherical Torus at closed flux surfaces and a force-free screw pinch (SP) at open flux surfaces and fed by electrodes [1]. By replacing the metal centrepost current of the spherical tokamaks with the SP plasma electrode current, the rod at the centre of the plasma, which represents the most critical component of spherical tokamak design configuration, can be eliminated. In this way the aspect ratio can be decreased during the experiment, meanwhile the ratio between the toroidal plasma current and the plasma electrode current, is increased. In order to verify the stability of the configuration and the technical components of the PROTO-SPHERA initial arc, a prototype facility has been constructed; this last experiment has reached its plasma current target in the recent months.
A diagnostic proposed to equip PROTO-SPHERA for the density measurement, that is of the order

of ~10e20 m−3, is a two color interferometer [2]. It will allow to obtain integrate density line in the equatorial plane, necessary to characterize this uncommon plasma. The design and the interferometer laboratory test are described in this paper.
[1] F. Alladio et al., Nucl. Fusion 46 (2006) S613–S624
[2] J. H. Irby et al., Rev. Sci. Instrum. 59 (1988) 1568

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Development of Winding Technology for ITER PF6 Double Pancakes

Co-authors: Wei Wen [1]; Shuangsong Du [1]; Guang Shen [1]; Weiyue Wu [1]; Jin Chen [1]; Kevin Smith [2]; Carlo Sborchia [2]; Peter Readman [2]; Yuntao Song [1]; Huan Wu [1]

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The Poloidal Field(PF) coils are one of the main sub-system of ITER magnets. The PF6 coil is being manufactured by the Institute of Plasma Physics, Chinese Academy of Sciences(ASIPP) as per the Poloidal Field coils cooperation agreement between ASIPP and Fusion for Energy(F4E).

ITER PF6 winding pack is composed by stacking of 9 double pancakes. Series double pancakes are being wound in ASIPP with a "two-in hand" configuration. This paper focus on the main winding process and results of ITER PF6 double pancakes. The winding workshop is composed by two symmetric winding line. During each double pancake winding, two conductors were simultaneously despooled, straightened, ultrasonic cleaned, sandblasted and then bent to the correct radius. Followed by manual cleaning, the conductors were wrapped with turn insulation by automatic wrapping head. Finally the conductors were accurately deposited onto the rotary table. 0.05% conductor forwarding length measurement and ±0.5mm radial build-up for each turn were achieved, which indicated well winding controlling. By now, 6 out of 9 ITER PF6 double pancakes winding has been successfully accomplished .

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An Optimization Study for Shielding Design of D-D and D-T Neutron Generators

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The neutron generator (NG) is being used more increasingly in various industrial and research area such as neutron activation analysis, neutron radiography, neutron capture therapy, and so on. In such an application of neutron generator, compactness is one of the most important issue. Since neutron source is generated by deuterium-deuterium (D-D) or deuterium-tritium (D-T) fusion reaction, relatively thick shield for both of fast neutron and related photon is usually required.
In this study, optimization of shielding design for portable D-D or D-T neutron generators was investigated by adopting appropriate moderator and shielding materials. The effects of moderator or shield materials to neutron spectra and dose rate were also studied to achieve minimization of shield. The polyethylene showed good performance in shielding D-D neutrons source while the tungsten showed considerable performance in shielding D-T neutrons due to relatively high threshold energy of (n,2n) reactions. After neutron shield, appropriate photon shield, such as the lead, was also required because the photon shielding performance of the polyethylene or the tungsten is not sufficient.
Final required shield thickness was also evaluated for various strength of D-D or D-T neutron

generators and maximum neutron source strength for portable neutron generator was also discussed.

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Environmenta,l steam-ingress and gamma irradiation tests for optical materials candidates for ITER equatorial-Vis/IR-WAVS diagnostic

Co-authors: Angel Ibarra [1]; Piedad Martin [2]

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To validate the design of Equatorial Vis/IR WAVS (Wide Angle Viewing System) diagnostic it is necessary to test materials and coatings of optical components and to characterize their behavior under the harsh environment of ITER (in particular in terms of radiation and temperature). The objective of this work is to summarize the results of the different tests: environmental (temperature, air, vacuum, steam-ingress) and gamma irradiation, carried out in different materials (candidates for port plug and optical hinge mirrors, vacuum window, field lens and beam splitters), in order to provide information for choosing relevant optical materials for this diagnostic.

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Neutronic effects of the ITER Upper Port environment update in C-Model

Co-authors: Arkady Serikov [1]; Luciano Bertalot [2]; Ulrich fischer [3]; Julio Guirao [2]; Eduardg Polunovskiy [4]; Raul Pampin [5]

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The goals of this work are the neutronic modeling of the ITER Upper Port (UP) environment according to the updates of ITER CAD model, the assessment of neutronic effects caused by that update and proposing improvements of the radiation conditions. The update has been applied to the ITER-reference neutronics simulation model called "C-Model" which includes the standard components and generic port plugs. The modifications of UP environment are related to the Vacuum Vessel (VV) UP extension and the port duct. The particular components of the UP environment are the following: VV double and single walls around the Diagnostic Generic Upper Port Plug (DGUPP), UP stub and blanket manifolds, In-Vessel coils (IVC) including Edge Localized Mode (ELM) coils and feeders, and vacuum sealing flange. The neutronics analysis of the newly developed model identified the components which influence predominantly the radiation field at the UP Inter-Space Structure (ISS) where personnel access is anticipated after shutdown. The radiation environment was analyzed in terms of neutron fluxes and Shut-Down Dose Rates (SDDR). High-resolution distributions of the SDDR have been obtained with the R2Smesh method of KIT coupling MCNP radiation transport simulations with isotope inventory calculations using the FISPACT activation code. The material specifications of the model are based on the approved material compositions taking into account the radiological impurity requirements. It improves the reliability of the radiation transport simulations for the assessment of the radiation conditions in the UP area and for the implementation of the ALARA principle for SDDR analysis. In conclusion, several design solutions are proposed to reduce

radiation streaming in the direction of the ISS maintenance area.
Disclaimer: This work has been funded by the ITER Organization (IO) service contract IO/17/CT/4300001478. This paper does not commit the IO as a nuclear operator. The model qualification is provided independently of this paper.

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Manufacturing and testing of flat type small size tungsten PFC mock-ups by HIP process

Co-authors: Eunnman Bang [1]; Heekyung Choi [1]; Hyoung Chan Kim [1]; Kyungmin Kim [1]; Suk-Ho Hong [1]

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W/CuCrZr PFCs will be used in ITER divertor and are strong candidate for the use in high heat flux regions of the upgraded KSTAR and K-DEMO. Development of hot isostatic pressing (HIP) bonding technology is in progress for the fabrication and qualification of tungsten divertor. We manufactured the first W/CuCrZr flat type small size mock-ups by HIP technology using PVD for interlayer formation. In the developed HIP process, we applied PVD method for interlayer coating of high purity Ni and Cu. The interlayer coating is selected to enhance the diffusion between W and Cu during the HIP bonding process. The Ni and Cu were deposited by two kinds of thicknesses on W surface. The bonded PFC of Ni/Cu coated W, copper interlayer and CuCrZr were manufactured by means of HIP technology employing two kinds of processing conditions. We observed the microstructure of the deposited Ni and Cu layer on the bonding interface by FE-SEM. The HIP processed samples were subjected to shear strength test and high heat flux (HHF) test. There is no significant difference in shear strength due to PVD coating thickness. But the shear strength of samples fabricated by 70 MPa and 700 ℃ HIP condition is higher than those fabricated by 60 MPa and 600 ℃ condition. HHF tests were performed to compare the thermal conductance through the bonded layer and thermal fatigue behavior of the samples in the tested condition as a preliminary test of the HIP process parameters. The layers between W and CuCrZr of samples were observed and compared by FE-SEM before and after HHF test. And bonding with Ti interlayer is compared to those of Ni interlayer in bonding strength and stability. All these results were analyzed synthetically to establish the optimum fabrication conditions and to find a method for performance qualification.

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A horizontal powder injector for W7-X

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Injection of low-Z powders into fusion plasma has been used to improve wall conditions, similar to the standard boronization process using diborane. Powder injection has the advantage of being much simpler, non-toxic, and efficient. The W7-X stellarator is planning on utilizing powder injection in long pulse discharges; a proof-of-principle test for horizontal injection into the plasma was conceived. A design concept is developed using a polyetheretherketone (PEEK) paddle wheel that is driven by a piezo motor, due to the high magnetic fields, that rotates at 100 deg/s. This small unit ("flinger") fits into an envelope of 120 mm diameter x 150 mm long, a standard size for Multi-Purpose Manipulator. The device is housed in a carbon cup mounted on a retractable probe that can be placed near the plasma edge, enabling powder injection ~4-8 cm radially into the boundary plasma. The feed for the paddle is via piezo electric actuator that vibrates a funnel filled with powder into a trough for the paddle to push. The 8 paddle arms, 35 mm long and 10 mm wide, are made from 0.38 mm thick PEEK which drag slightly along the powder-filled trough bottom, becoming a spring-loaded paddle which accelerates the powder upon release. Design challenges are the high ambient magnetic field, vacuum compatible materials, high temperature environment, limited rotary-drive options, and compact space. The design and testing of this new device will be presented.

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Assessment of environmental effects on the ITER FOCS operating in reflective scheme with Faraday mirror

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Plasma current measurements will play an important role in ITER to provide real-time plasma control and machine protection. Fiber Optics Current Sensors (FOCS) with the sensing fiber installed on the external surface of the vacuum vessel is a system intended to perform this task. The FOCS signal is proportional to the current and is more suitable for the steady-state operation as compared to today's standard electromagnetic sensors. However, the intrinsic and extrinsic linear birefringence significantly degrade the sensor performance. Recent numerical simulations demonstrated that operating FOCS in reflection with an ideal Faraday mirror (FM), i.e. the rotation angle is exactly 90°, can significantly improve the measurement accuracy. Unfortunately, FM detuning above 0.3° gives unacceptable error, while the tolerance of commercial FMs is ±0.5°, at best. The FM calibration offers only a partial solution due to uncontrolled detuning of the FM under external perturbations. This problem is the subject of the presentation.
An ideal FM can be located far from the vacuum vessel (in the cubicle area) because it fully compensates the linear birefringence in the link between the sensing fiber and the FM. For a non-ideal FM this link gives additional errors and should be avoided. When located near the vacuum vessel the FM will be exposed to thermal and radiation fields which will influence the FM rotation. We have investigated these effects experimentally and found that the thermal detuning sensitivity is ~0.12°/K, i.e. temperature stabilization better than ±2.5K is necessary. Radiation-induced detuning saturates at ~1.2° for gamma-doses of 12.5kGy@520Gy/h. Saturation indicates that pre-irradiation may be a way of radiation hardening.
In the presentation we will also discuss the physical background for the temperature and radiation sensitivity and show that our experiential results agree well with the theoretical models assuming that the FM uses Bismuth-Iron Garnet as the Faraday rotator.

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The helium turbo circulator – the heart of cooling systems

Co-author: Martin Kroupa [1]

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In few last decades, great attention was paid to development in the field of fusion technology. Currently, the International Thermonuclear Experimental Reactor (ITER) is under construction followed by Demonstration Power Station (DEMO) which should be first nuclear fusion power plant in the world. Both of these facilities have one point in common – high power density and thus great demands to supporting cooling systems. Therefore, hand in hand with development of fusion technology, it is also necessary to pay attention to cryogenic, refrigeration and cooling technologies. The paper presents development of helium Turbo Circulator (TC) installed in the form of two pieces in experimental testing facility HElium LOop KArlsruhe (HELOKA) in Germany and as one piece in HElium For FUSion (HEFUS-3) located at ENEA Brasimone, Italy. All installed TCs have output power 232 kW and were/are used in pressured helium circuits for cooling mock-ups of important ITER parts – Test Blanket Module (TBM) and Tested Diverter Module (TDM). Simultaneously, the paper summarizes gained experience and deals with a concept of bigger TCs which could be used in real cooling systems for cooling Inboard Blankets (IB), diverters and Outboard Blankets (OB) of fusion reactor in ITER and DEMO project.

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Design and measuring performance of the ITER plasma position reflectometer in-port-plug antennas.

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Design parameters of the ITER Plasma Position Reflectometer (PPR) in-port-plug antennas are determined and then their measurement performance is assessed using 2D full wave analysis.
Two ITER scenarios were selected when considering the optimum antenna position and orientation, namely the baseline scenario (15 MA D-T) and the low density one planned for the initial non-active phase at 7.5 MA. Using them to feed a 3D ray tracing simulation, spatial position and optimum orientation angles of each set of emission and detection antennas were determined. Additionally, a far field analysis of the launching radiation patterns led to the definition of the antenna dimensions in terms of optimal power coupling.
After this preliminary work, 2D full wave simulations using a finite difference time domain (FDTD) code were performed to assess the measurement performance of the system, in terms of spatial resolution and accuracy. To this end, the detected wave amplitudes and phases were evaluated for each operating scenario and two models of the SOL plasma density. By calculating the spectrogram (STFT technique) of the phase of the detected wave, the reconstruction of the plasma density profile can be carried out and be directly compared with the input profiles. As a result from this synthetic diagnostic analysis, an estimation of the error in the determination of the last closed flux surface position was possible. Both static and turbulent plasmas have been considered, using for the latter a multimodal model to mimic the spectrum of density fluctuations.
Together with the power gain information previously obtained, the 2D characterization of the measuring performance of the system will be presented.

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Fault analysis and improved design of JET In-vessel Mirnov coils

Co-authors: Matteo Baruzzo [1]; Giovanni Artaserse [2]; Rafael Henriques [3]; Sergei Gerasimov [4]; Norman Lam [4]; Peter Lomas [4]; Ruben Otin [4]; Fernanda Rimini [5]; Maximos Tsalas [6]; Steven. Van Boxel [4]

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In vessel Mirnov coils are an essential diagnostic in present day tokamaks. Their use in ITER and future Fusion reactors presents some disadvantages linked to the high radiation environment. Furthermore large Electro Magnetic forces can be experienced by the coil, due to the pulsed operation of the tokamak device [1], and disruptions [2].
Since the operation with the ITER-like wall, JET has experienced severe faults in the high-bandwidth Ti wire coils. Surface discoloration of the wire pointed to the formation of chemical surface layers that can produce hardening of Ti, leading to increased mechanical failures. Disintegration of 0.5mm thick mica plates covering slots in the coils casing was also found.
The tensile strength of the failed wire has been tested, comparing it to new samples of the same composition and manufacturer, and hardening has been observed. Scanning Electron Microscope investigations have been carried out on the rupture surface, which is flat but rough, with no wire deformation, compatible with fatigue failure. Modelling of EM loads showed that the forces, combined with fatigue on hardened Ti are enough to cause failure on sharp wire bends.

During 2016-17 new coils have been designed and installed. These can be replaced using remote handling, and they use Cu alloy wire, in order to reduce chemical interaction with in-vessel gases and increase heath conduction to the ceramic former.
The presented work includes the failure analysis and modelling, motivating the design differences between old and new coils. The latter will provide valuable information on the long term effects of EM loads during disruptions, as well as chemical degradation processes that will be encountered for ITER HF coils, which are characterized by the same materials.
1) Vayakis, George, et al. Journal of Nuclear Materials 417.1 (2011): 780-786.
2) Gerasimov S.N. et al Nucl. Fusion 55 (2015) 113006

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Post-Mortem Analysis of ITER CS Helium Inlets Fatigue Tested at Cryogenic Temperature

Co-authors: Ignacio Aviles Santillana [1]; Stefano Sgobba [1]; Cornelis Jong [2]; Paul Libeyre [3]; Sandra Castillo Rivero [1]; David Everitt [4]

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In the ITER Magnet System, ten thousand tonnes of superconducting cable – in – conduit - conductor (CICC) are cooled down by a forced flow of supercritical helium, which is supplied from helium inlets. For the ITER Central Solenoid (CS), consisting of six independent pancake wound modules, the He inlets consist of three overlapping holes covered by an oblong shaped boss, welded to the CS jacket through full penetration, multi-pass Tungsten Inert Gas (TIG) welding. There are 120 CS He inlets and because they are located in a region of high cyclic tensile stresses, i.e. first turn at the inner diameter of the pancake, the CS inlets are one of the most critical structural components.

Qualification of the design is done by analysis and a comprehensive design optimization has been performed by finite element (FE) analyses. In order to guarantee the required fatigue life at cryogenic temperature of these component, a post – welding process consisting in ultrasonic shot – peening is required.

Based on a qualified weld procedure, six mock – ups including each two He – inlets on the opposite surfaces have been produced to run a mechanical fatigue testing program at cryogenic temperature with sufficient statistical significance to validate the findings of the FE simulations. Five were peened. One not peened.

The paper describes the results of a comprehensive post – mortem failure analysis which includes non – destructive (penetrant testing, leak testing, computed tomography) as well as destructive examinations (microoptical and hardness tests, scanning electron microscopy). It also includes a full assessment of the welds according to the most stringent acceptance levels of the standards in force.

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

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Enhanced Droplet Control for the Fabrication of Ceramic Breeder Pebbles

Co-author: Oliver Leys [1]

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Lithium rich advanced ceramic breeder pebbles composed of lithium orthosilicate with a strengthening phase of lithium metatitanate are intended as tritium breeders for future fusion reactors. The EU breeding blanket being designed for trial in ITER will feature the pebbles in the form of pebble beds in the wall of the reactor. Upon irradiation with neutrons, the lithium will decay into tritium and helium, after which the tritium is processed before being rerouted into the reactor core to react with deuterium and completing the fuel cycle. It is imperative that the pebbles are of sufficient quality to ensure the smooth functioning of the breeder blanket.
At the Karlsruhe Institute of Technology, a melt based process is used to produce the pebbles. A melt is formed in a platinum alloy crucible from synthesis powders at approximately 1400 ℃. Upon ejection through a 300 µm nozzle, a laminar jet is formed which breaks up into small droplets due to Plateau-Rayleigh instabilities. These describe how random instabilities grow on the surface of the jet until the surface tension overcomes the viscous forces, causing droplet break-off. However, as the break-up is considerably irregular, small changes in the droplet sizes will result in droplets with different momentum causing more to coalesce, resulting in oversized pebbles.
In order to control the break-up of the jet, and indirectly the size of the pebbles, instabilities in the form of audio waves were deliberately applied to the system. A high-speed camera was used to study the effects of varying frequencies on the behaviour of the jet. By applying the correct frequency, it was possible to increase both the yield as well as the monodispersity of the product by minimising the number of droplets coalescing, resulting in less waste and more control of the end product.

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Radiolysis study of EU Li4SiO4 reference breeder material from the HICU experiment

Co-author: Julia Heuser [1]

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To proceed the solid breeder concept for ITER and DEMO it is essential to investigate Ceramic Breeder (CB) materials' properties. To ensure an adequate tritium production of the breeder material several requirements like a high lithium density, good tritium release behaviour, and a high resistance against neutron irradiation as well as thermomechanical stresses have to be fulfilled. Lithium orthosilicate (Li4SiO4), applied as pebbles, has been selected as reference material in the European Helium Cooled Pebble Bed (HCPB).

Beside standard material characterization, the response of CB materials to neutron irradiation is an important issue. Therefore, CB pebbles of the EU reference material were exposed to neutron irradiation in the HICU experiment (high neutron fluence irradiation of pebble stacks for fusion), that was carried out in the High Flux Reactor (HFR) in Petten (Netherlands) between 2008-2010. Different grades of Li4SiO4 pebbles containing a surplus of 2.5 wt% SiO2 and different 6Li-contents up to 20 % were included in the irradiation under DEMO relevant conditions.

While the Post-Irradiation Examination (PIE) on tritium release behaviour and material properties were recently presented, new and additional results of a radiolysis study on the Li4SiO4 samples was performed in Latvia will be presented here. Radiation induced Defects (RD) and Products (RP) were investigated using basically Electron Spin Resonance (ESR) and Raman spectroscopy. Further analyses on the tritium release behaviour were performed using Thermally Programmed Desorption (TPD).

The presented results will reveal new insights of CB pebbles' behaviour with regard to neutron irradiation and will therefore significantly contribute to the knowledge of CB pebbles' properties in a fusion relevant environment.

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SMITER: a field-line tracing environment for ITER

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The ITER plasma-facing components (PFC) are now fully designed and procurement is underway. A key utility in such design is field line tracing for different magnetic equilibria which allows the definition of component front surface shaping. On ITER, this design phase has deployed both analytic theory [1] and the tracing codes CASTEM and PFCFLUX [2]. Attention is now turning towards the critical issue of management and control of PFC heat fluxes, so important in an actively cooled device such as ITER. To facilitate the development of control algorithms, particularly for protection of the beryllium main chamber first wall panels [3], a new field line tracing environment, SMITER, has been developed, featuring a sophisticated Graphical User Interface (GUI) that uses the SMARDDA [4] kernel and has been thoroughly benchmarked against PFCFLUX for specific cases of first wall panel and divertor target loading. SMITER allows power deposition mapping in the full 3-D CAD geometry of the machine, taking as input a user-defined specifications for parallel heat flow in the scrape-off layer. In addition to its use as input for shape control algorithms [3], the tool will also be important for the production of synthetic surface temperatures using built in thermal models for input to diagnostic design.

The newly developed GUI framework provides CAD and IMAS integration, parametric CAD components catalogue, meshing, visualization, Python scripting, storage in hierarchical data files (HDF) with several simulation cases in one study running in parallel and using MPI for code speedup. Integrated ParaView module can augment CAD geometry, meshes and results.

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Isotope separation systems for a european DEMO

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Large-scale isotope separation in a DEMO tritium plant poses significant challenges. Alternatives to distillation and palladium-based adsorption (used in the tritium fuel cycle for JET) remain elusive, despite the disadvantages: Cryodistillation is energy intensive and lacks inherent safety due to the high tritium inventories in the liquid phase, that inevitably expand to vapour in the event of a cooling failure. Palladium-based packings are expensive, and the systems are limited in scale by the heat transfer required to liberate hydrogen from the metal hydride. This presentation will outline the approach taken to tackle these challenges by the European DEMO Tritium Matter Injection and Vacuum work package, and highlight the most interesting results.

Firstly, a candidate list of potential technologies was generated. The isotope separation requirements created by each system block in the fuel cycle was considered, and potential mass balances for different scenarios defined. This work has identified approaches to the plasma fuelling requirements which could reduce isotope separation requirements, and opportunities to integrate isotope separation requirements within the fuel cycle.

The combination of requirements and mass balances has then been used to assess each technology option qualitatively and, where possible, quantitatively. This has led to surprising results in the size of systems required and the feasibility of both older and novel separation technologies for two distinct sets of isotope separation requirements: isotope rebalancing and protium removal for the inner fuel cycle, and trace tritium recovery for the outer fuel cycle. Studies have established that cryodistillation and temperature swing adsorption type processes (such as TCAP) are feasible within the inner fuel cycle, while gas chromatography is less attractive. Work is on-going to assess thermal diffusion, membrane separations, and pressure swing adsorption, and the suitability of technologies for trace tritium recovery in the outer fuel cycle.

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Progress of the EU activities for the ITER Divertor Inner Vertical Target procurement

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F4E undertook the qualification of so-called "Additional Suppliers" in order to enhance competition among the potential bidders and secure the procurement of the ITER Divertor Inner Vertical Target. In order to assess the performances of W armoured Plasma Facing Components under the conditions expected in the divertor target strike point region, a total of 36 W monoblock mock-ups were manufactured by ATMOSTAT (F) by Diffusion Bonding, and by CNIM (F) and Research Instruments GmbH (D) by brazing. 24 mock-ups were High Heat Flux (HHF) tested in IDTF (Efremov Institute Saint Petersburg, Russian Federation) electron beam test facility.
The HHF testing program foresaw the performance of 5000 cycles at 10 MW/m2 and 300+700 cycles at 20 MW/m2 with 10 s power on and 10 s dwell time. The coolant conditions were representative of the Inner Vertical Target ones and a swirl tape (twist ratio = 2) turbulence promoter was provided.
The test results showed some significant improvements, in particular the issues of W monoblocks macro-cracking and heat sink thermo-mechanical fatigue performances, identified during previous HHF tests campaigns. Some critical heat flux experiments were also performed.
The main results will be presented and discussed in the paper.

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Hydraulic characterization of twin box joints for ITER magnets

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The ITER magnet system will be the largest superconducting magnet system ever built. The system, all inside a cryostat, is mainly composed by a central solenoid (CS) split in 6 modules, a set of 18 toroidal field (TF) D-shaped coils and 6 poloidal field (PF) coils. Each of these coils use variable type of cable-in-conduit-conductors (CICC) actively cooled by supercritical helium forced flow. Their electrical supply from the current feedthrough of the cryostat is done with main busbars (MB) using similar CICC. The electrical MB to coils as well as internal PF and TF coils connections rely on the twin box concept developed by CEA in the early R&D phase. After construction and electrical validation of joint prototypes for the PF and the MB conductors through full size samples, the question of their hydraulic behavior in the operating conditions arises. Two specific hydraulic characterization tasks were done through the Magnet Infrastructure Facility for ITER (MIFI) contract between ITER

Organization (IO) and CEA devoted to develop, improve and qualify manufactured components and assembly processes. These support tasks, were done on the full size qualification samples by setting the samples on the CEA OTHELLO dedicated facility able to operate with gaseous N2 in a large Reynolds range at room temperature. The paper explains the way followed to get a full hydraulic characterization of the MB and PF5 half joint boxes. The pressure drop for the two flow directions was determined for both joints. The study of the flow distribution between parallel cooling channels inside the PF5 joint revealed a bypass of the active joint region. The paper reports on this hydraulic behavior in the relevant magnets operating conditions and outlines the design changes in the joints provoked by the results of this study.

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A preliminary assessment of MCNP unstructured mesh integration in the ITER neutronic model

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The design of nuclear fusion devices like ITER requires the execution of complex multi-physics simulations, involving different analysis disciplines such as mechanical, thermal-hydraulic and neutronics. Nowadays, thanks to the novel implementation of unstructured mesh capability into MCNP6, nuclear responses can be computed over meshes conformal with the components tackling the problematic load transference between different codes and, at the same time, improving the modelling methodology.
Initially a sensitivity analysis over type, order and number of elements was performed to evaluate the hybrid geometry performances in terms of memory demands, time of loading and the different MCNP implementation methodologies.
Subsequently, a limited section of the ITER vacuum vessel was modelled with unstructured meshes by means of the HyperMesh code. This region was chosen as one of the examples where the explicit structured mesh tally scoring can produce unphysical results near the cells boundaries where the mesh tally voxels cross two (or more) cells with different material properties and relative cross sections. Results are compared with the standard Constructive Solid Geometry (CSG) ITER model and discussed. Samplings of the unstructured mesh tallies were performed and locally assessed against the structured tallies. The particle fluxes and nuclear heating maps obtained show an overall agreement on the average resulting values, however significant local deviations were observed. In particular, unstructured mesh local results showed higher degree of correlation to the underlying materials and avoided the presence of unphysical peaks. However, the added complexity and the time required for meshing large regions, which include many complicated parts, makes an analysis of the benefits necessary, and it limits the application of hybrid geometries to particular cases.

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3D Tritium Transport Model at Breeder Unit Level for WCLL Breeding Blanket

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The Water-Cooled Lithium Lead (WCLL) Breeding Blankets is one of the European blanket designs proposed for DEMO reactor. A tritium transport model inside the blankets is necessary to assess their preliminary design and safety features. Tritium transport and permeation are complex phenomena to be taken into account in the evaluation of tritium balance in order to guarantee tritium self-sufficiency and to characterise tritium concentrations, inventories and losses. In this context, the study has been performed at breeder unit level in the outboard equatorial breeding blanket module,

which is, during the normal operating conditions, one of the most loaded modules and this results in higher permeation phenomena. For these purposes, a 3D transport model has been investigated; the model includes buoyancy effects and a preliminary evaluation of the magneto-hydro-dynamics effect (MHD) is also performed. Moreover, it includes advection-diffusion of tritium into the lead-lithium eutectic alloy, transfer of tritium from the liquid interface towards the steel (adsorption/desorption), diffusion of tritium inside the steel, transfer of tritium from the steel towards the coolant (recombination/desorption), advection-diffusion of diatomic tritium into the coolant and buoyancy effect. The temperature profile, tritium generation rate profile, and Pb-15.7Li flow velocity profile have been also taken into account. Results deriving from the transport equation solution, with the above specified phenomena, input and boundary conditions are illustrated in detail within the paper.

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Numerical study of conjugated heat transfer for DONES high flux test module

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Helium flows at low pressure (0.3 MPa) are used to cool the specimen capsules and the structure of the neutron irradiated High Flux Test Module (HFTM) of the DEMO-Oriented Neutron Source (DONES). The flow path includes inlet and outlet ducts with large cross sections, but also mini-channels with 1 mm gap width, where a high velocity low Reynolds number laminar to turbulent transitional heated flow influences the temperature of the irradiated specimens. The large span of Reynolds numbers from laminar to fully turbulent are a significant challenge for the simulation of the complete HFTM and requires validation of models.
Numerical simulations have been carried out for turbulent (Re = 4500, 6000 and10000) helium flow in a heated mini-channel using the commercial CFD code Star-CCM+ v.12.06. The results were compared with measurements obtained from the IFMIF thermal-hydraulic experiments (ITHEX). Detailed comparisons were made between the k-ϵ, k-ω-SST, and V2F models with different treatment methods of near-wall layer. The appropriated models have been used for the numerical thermo-hydraulic analysis of the conjugated heat transfer the in HFTM. The temperature distribution in the HFTM structure calculated for normal operational conditions serves as a basis for a subsequent thermo-mechanical structure analysis of the HFTM.
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Nuclear analyses of solid breeder blanket options for DEMO: status challenges and outlook

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Within the Power Plant Physics and Technology (PPPT) programme of EUROfusion, an intensive development effort is devoted to the detailed design of a solid breeder blanket for a demonstration fusion reactor (DEMO) with the inherent capability of a highly efficient tritium breeding. A novel design of the Helium Cooled Pebble Bed (HCPB) breeding blanket based on a Single Module Segment (SMS) and an enhanced, near term configuration has been developed for the EU DEMO reactor, having high flexibility with regard to tritium generation and global reactor performances. The key part of the analyses is the development of the geometry model of the breeder blanket with enough detail

to perform high fidelity nuclear simulations. To this end a generic geometry model of a DEMO 2017 baseline was adopted to arrange the newly developed SMS HCPB layout using a novel modelling technique. Separate models were developed using detailed CAD designs and the McCad conversion tool: the SMS blanket module, a breeder unit and a highly detailed First Wall (FW) with cooling channels and a roof-top shape. In this way, the model includes a detailed representation of the breeder zone, the back supporting structure and the FW of the blanket.
This model has been applied for the study of this new generation of HCPB blanket , as well as for the investigation of an alternative concept based on liquid lead neutron multiplier, replacing the Be/beryllide one. The simulations include assessment of the main required nuclear responses: TBR, nuclear power generation and shielding performances. A special technique has been also applied to find the optimal geometry configuration of the different concepts. The study concludes with sophisticated activation analyses of the in-vessel components and with the discussion of different effects coming from the detailed modelling and affecting engineering solutions, current design highlights, challenges and outlook.

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A 15 T large aperture dipole for testing fusion and accelerator superconducting samples

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High field superconducting magnets are an essential technology enabling the development of magnetic confinement fusion and high energy hadron colliders. These two communities have joined efforts to design a facility for testing superconducting cables and small insert coils. We propose a large aperture Nb3Sn dipole to replace the magnet assembly of EDIPO, which was irreversibly damaged in 2016, while the rest of the EDIPO infrastructure, including cryostat, cryoplant, power supplies, and high current transformer, remained intact. The goal of this new magnet is to generate a background field of 15 T over a ±0.5% homogeneous field length of 1000 mm. The aperture will enable the test of samples from the two test facilities existing worldwide for the test of high current superconducting cables in a background field above 10 T: SULTAN (94×144 mm aperture) and FRESCA2 (100-mm-diameter bore). The main features of the magnet design will be discussed along with the status and outlook of this collaborative effort between Fusion and High Energy Physics laboratories.

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Neutronic assessment of HCCR breeding blanket for DEMO

Main goals of breeding blanket development in Korea are to develop and verify the integrated blanket design tools; to develop the core technologies such as blanket materials, blanket cooling, and tritium fuel cycle technologies; and to develop and evaluate fabrication and joining technologies. Several breeding concepts are considered as candidates for the Korean DEMO blanket concept. As a solid

breeder concept, helium-cooled ceramic reflector (HCCR) blanket and water-cooled solid breeder blanket (WCSB) are considered. The HCCR blanket adopts the unique graphite reflector concept to reduce the amount of beryllium multiplier with the Li2TiO3 breeder, reduced activation ferritic-martensitic steel structural material, and helium coolant. This concept of HCCR blanket is adopted to be tested in ITER as a Test Blanket Module (TBM). Currently, the design and R&D activities for HCCR blanket has been performed in Korea.
In this paper, the design concept of the HCCR breeding blanket system is explored to satisfy the global TBR requirement by a neutronic assessment based on the K-DEMO neutronic analysis model employing vacuum vessel, toroidal field coil, blanket and shield. Firstly, sensitivity studies were performed for a various combination of HCCR blanket breeding layers with fixed blanket thickness. The thicknesses and arrangement of the breeder, multiplier and reflector layers in the HCCR blanket were optimized in the view of tritium breeding ratio. Secondly, the neutron flux distribution inside the blanket was calculated to evaluate the neutron shielding performance of the blanket. Finally, the nuclear heat distribution on the blanket was also estimated to support the cooling system design study.

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Surface oxidation effect on deuterium permeation in reduced activation ferritic steel F82H for DEMO application

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Tritium permeation through structural materials is a significant issue for the Japan's DEMO reactor blanket concept. Reduced activation ferritic steel F82H is a prime candidate for the blanket structural material. The previous study showed a thin chromium oxide layer formed on a steel substrate worked as tritium permeation barrier; however, heat treatment parameters at atmospheric pressure for the formation of a chromium oxide layer and its permeation behavior in DEMO reactor environments are not clear. In this study, surface oxidation treatments for F82H have been performed to evaluate the effect of the chromium oxide layer on deuterium permeation and its stability under actual DEMO reactor conditions.
To form a tight chromium oxide layer without formation of iron oxide which causes cracks in the layer, F82H steel plates were heat-treated under low oxygen partial pressure conditions: for 5 min at 700–720 °C in gas flow of argon-hydrogen mixture. After surface observation and analysis, gas-driven deuterium permeation measurements were performed at 300–600 °C. The selected samples were exposed to purge gas of helium with 1 vol% hydrogen or liquid lithium-lead for 100 h at 500 °C for the simulation of DEMO blanket conditions.
The optimized heat-treatment parameter for chromium oxide formation without iron-oxide formation was determined: for 5 min at 710 °C in half-and-half argon-hydrogen mixture gas. The thickness of the layer was estimated to be less than 100 nm. Deuterium permeation flux of the sample decreased by a factor of 10 in comparison with untreated F82H in the first measurement at 300 °C, and showed a further decrease by a factor of 150 at 500 °C due to an increase in the layer thickness by 1.5 times during the permeation measurements. However, the chromium oxide layer was lost with an increase in deuterium permeation after exposure to DEMO blanket environments.

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Improving a Negative Ion Accelerator for next generation of Neutral Beam Injectors: results of QST-Consorzio RFX collaborative experiments

In large Neutral Beam Injectors for fusion applications, the efficiency of ion beam neutralization and

transport to the tokamak plasma strongly depends on the divergence and the deflection angle of each single beamlet with respect to its ideal trajectory. In fact, a very narrow window is available for the particle beam to pass through the neutralizer panels and the duct reaching the tokamak plasma. For this reason, beam optics quality is one of the key requirements in multi-stage multi-beamlet negative ion accelerator, such as the high power Heating Neutral Beam injectors for ITER and JT-60SA.
In the framework of the collaboration established between Consorzio RFX (Padova, Italy) and QST (Naka, Japan) experimental campaigns have been organized on the Negative Ion Test Stand (NITS) in Naka employing an ITER-like multi-beamlet configuration [1].
These campaigns were intended to:
• Test and optimize new solutions for cancelling the undesired ion deflection caused by the transverse magnetic field required for suppressing the co-extracted electrons [2].
• Better understand the physics underlying the extraction of single beamlets in the initial part of their trajectories, close to the "meniscus" region.
• Validate and improve the numerical codes used for the accelerator design
Although it was not possible to explore all the intended parameter space due to power supply limitations, the experiments carried out in 2016 and 2017 allowed to successfully achieve the first most important (engineering) objective and to provide bases for the achievement of the other two (physics) objectives, which will be completed during new already planned activities and collaborations.

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Optimal configuration of a tokamak fusion system with breeding blanket based on ITER TBM

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System parameters and the optimal radial build of a tokamak fusion system with a normal aspect ratio were found through the coupled analysis of a tokamak system and neutron transport. Neutron impact on shielding and tritium breeding capability are self-consistently incorporated together with plasma physics and engineering constraints in determining the radial builds. The plasma physics and engineering constraints moderately extrapolated from the constraints adopted in the design of International Thermonuclear Experimental Reactor (ITER) were used. In a tokamak fusion system with a normal aspect ratio, configuration with only an outboard breeding blanket does not satisfy requirement on tritium self-sufficiency. The optimum system size to produce a given fusion power was determined by the requirements on the shielding, tritium breeding and the magnetic field at the toroidal field (TF) coil. Thickness of the outboard and inboard blanket were determined to allow optimal system size. With a confinement enhancement factor H = 1.3, Q > 30 was possible for fusion power greater than 2,000 MW with an aspect ratio of A = 3.0; however, Q > 30 was impossible with an aspect ratio of A = 4.0. The tritium breeding capability of blanket concepts proposed for testing in the International Thermonuclear Experimental Reactor (ITER) was evaluated by varying the outboard/inboard blanket thickness and the degree of lithium-6 (Li-6) enrichment. Cases with a smaller aspect ratio exhibited better performance since the number of fusion neutrons that contributed to tritium breeding were larger than the case with a larger aspect ratio. Among the blanket concepts, a helium (He)-cooled solid breeder (HCSB) concept showed the best tritium breeding capability and thus allowed for a smaller system size.

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Experimental analysis of dummy load prototype for ITER coil power supply system

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This paper mainly introduces the experimental analysis of the dummy load prototype, whose functions are to verify the capability of the ITER magnetic power supply systems to operate at their rated power levels without energizing the superconducting coils. The rated inductance of dummy load is 6.73 mH and the pulse test currents are 45 kA, 55 kA and 68 kA. To meet the requirements of the large inductance and different pulse test current classes, a dry-type air-core water-cooling prototype with epoxy resin casting technique has been fabricated, which has 24 layers and 72 turns. The experimental analyses are introduced in detail, focusing on the consistency test, inductance test, and temperature rise test. The consistency of different coil layer is ensured by comparing the voltage wave under the same voltage excitation. The inductance test includes dc inductance test by using the first order circuit method and ac inductance test by LCR instrument and finite-element simulation analysis. The temperature rise test results under different current classes are introduced by comparing with the simulation analyses. The above experimental results are coincided with the requirements and it shows the availability and feasibility of the prototype.

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Preliminary RAMI assessment for ITER test blanket module ancillary systems

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A RAMI (Reliability, Availability, Maintainability and Inspectability) assessment performed on the ITER Test Blanket Module ancillary systems is presented. The assessment is aimed at evaluating design criticalities possibly jeopardizing the achievement of the overall 75% availability requirements for the considered ITER plant. The Ancillary systems of the European Test Blanket Systems for ITER here analysed are the helium cooling systems (HCSs) of both the Helium Cooled Lithium Lead (HCLL) and Helium Cooled Pebble Bed (HCPB), lithium-lead loop of the HCLL TBM and the Tritium Extraction System (TES) of the HCPB TBM.
Reliability and availability performance were assessed by means of reliability block diagrams (RBDs) over a foreseen mission of 11 days and 20 years respectively, with operating cycles reflecting ITER schedule. In particular, a set of events leading to unavailability of the systems was initially defined by means of a failure mode and effect analysis. Then RBDs were implemented according to reliability-wise integration of the components included in such systems. Different RBDs were defined depending on the component judged to impact normal operation or start-up operation mode.
Systems analysis was performed exploiting a modular approach in order to elicit the relative contribution of specific components to the system availability so to support possible design improvement by highlighting critical sub-systems. In particular the cooling, tritium extraction and trapping functions were assessed separately for HCLL and HCPB concepts.
Both HCLL and HCPB ancillary systems presented design resulted able to achieve the availability and reliability targets with performance ranging from 83% up to 99% for reliability at 11 days and from 89% up to 99% for mean inherent availability. Finally the integrated impact of all ancillary systems on overall tokamak operational availability was estimated and the assessed systems, also considering the current preliminary level of design development, appear able to match expected performance.

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Development of Experimental Helium Cooling Loop (EHCL) for testing nuclear fusion blanket components

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Institute for Plasma Research is developing an Experimental Helium Cooling Loop (EHCL) as a part of R&D activities in fusion blanket technologies. This helium cooling system is designed for testing various nuclear fusion components such as tritium breeding blanket, helium-cooled divertor, and any other components which can be operated within EHCL operating window. The cooling channels of breeding blanket and divertor comprise of complex channel geometry having several parallel channels carrying helium gas for efficient heat extraction. Several mock-ups of these systems need to be tested before finalizing the design and fabrication. In addition to the individual testing of mock-ups of breeding blanket and divertor, integrated operation of the loop as well as understanding the behaviour of high-pressure and high-temperature system components are very essential for the development of Helium Cooling System for a fusion reactor.

This paper discusses the preliminary process design, process & instrumentation details, and operating & design parameters of the EHCL system. It also describes the characteristics of the EHCL system and the mechanism used to control various loop parameters. It briefly discusses about the architecture of major subsystems and components of the loop. Performance test results of the circulator(s) with its associated loop are also presented in this paper.

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Characterization of modified Be13Zr beryllide as advanced neutron multiplier

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Hydrogen generation reaction with water vapor of Be at high temperatures and BeO produced by this reaction that is harmful to human bodies are major drawbacks. Advanced neutron multipliers with high stability at high temperatures are desirable for fusion reactors where coolant water is extensively used. Beryllides have strong potential for use in high-temperature environments. In the framework of Broader Approach (BA) activities, beryllide pebbles as the advanced neutron multiplier were successfully fabricated by a combination of a plasma sintering synthesis method and a rotating electrode granulation method (REM).
Beryllide disturbs the tritium breeding by the metal in Be such as Ti, V. Therefore, Be13Zr was selected, because Be13Zr not only has low neutron absorption property, but also has no peritectic reaction during granulation. Be13Zr pebble has been successfully fabricated directly by REM using the plasma-sintered Be-Zr electrode.
However, Be13Zr has pest phenomenon. Pest reaction occurs during oxidation at relatively low temperatures, and is disintegration of polycrystalline sample cased by oxidation. The disintegration into powdery products is occurred by the growth of oxide along the grain boundary or crack. As action on this issue, dispersion of insoluble element for reduction of oxidation reaction was tried. Si has been selected as dispersion element, because they have low solubility in Be and low neutron cross-section. From the optimization result, Be13Zr without pest reaction was successful in developing by addition of Si. It is supposed that pest reaction was prevented by reduction of stress of oxidation reaction caused by dispersion of Si in grain boundary of Be13Zr.
In the present study, the characterization of modified Be13Zr by the addition of Si without pest reaction will be introduced, such as thermal and chemical properties.

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MHD mixed convection flow in the WCLL: heat transfer analysis and cooling system optimization

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In the Water-Cooled Lithium Lead (WCLL) blanket, a critical problem faced by the design is to ensure that the breeding zone (BZ) is properly cooled by the refrigeration system, thus to keep the structural materials under the maximum allowed temperature. For this purpose, CFD simulations are carried over using ANSYS CFX to investigate how the cooling system performances are affected by the tokamak magnetic field.
In the configuration studied, the blanket relies on a Single Module Segmentation approach. The LiPb flows inside long poloidal channels, developing along the whole module height; cooling is provided by double-walled tubes which are separated by a vertical pitch and inserted in the BZ from the back-supporting structure.
The attention is focused on the sub-channel closest to the first wall (FW). The maximum of the neutronic power deposition is foreseen in this region; thus, intense buoyancy forces will arise due to the large temperature gradient in the radial direction (Gr≈10^10). These will interact with the forced convection flow (Re≈10^3), leading to the onset of a mixed convection regime. A constant magnetic field parallel to the toroidal direction is assumed with intensity in the Hartmann number range 0≤M≤10^4. The walls bounding the channel and the water pipes are modeled as perfectly conducting.
The magnetic field is found to dampen the velocity fluctuations triggered by the buoyancy forces and, for M=10^4, the flow can be reduced to a pure forced convection regime. This effect causes the sharp reduction of the LiPb heat transfer coefficient to one-third of the ordinary hydrodynamic value. Consequently, hot-spots between the nested pipes and close to the FW are observed where the temperature exceeds the maximum allowed value. To address this issue, optimization strategies for the BZ cooling system layout are proposed and implemented in the CFD model, fulfilling the design criterion.

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Supplementary neutronics analysis of DEMO WCLL including activity and decay heat

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The WCLL (Water Cooled Lithium Lead) is a European option of the breeder blanket dedicated for DEMO fusion power reactor as being developed in the frame of EUROfusion's Power Plant and Technology (PPPT) programme. The intense neutron radiation produced results in a strong activation of the breeder blanket structural elements. The activation and decay heat generation of the WCLL components need to be assessed for maintenance, decommissioning and waste management purposes and related safety analyses.
This paper presents the analyses performed within the SAE (Safety and Environment) project of EUROfusion/PPPT aimed at providing up-to-date estimates of the activity inventories and the decay heat generation in the WCLL. A detailed investigation based on a set of coupled MCNP neutron transport and FISPACT inventory calculations were performed using the 2017 WCLL MMS (Multi-module Segment) model and the FENDL-3.1 nuclear cross-section data library. Activity inventories and decay heat data were assessed for the different breeder blanket segments to take into account heterogeneity.
The paper discusses the results obtained for the activity and the decay heat as a function of the decay time after radiation and also addresses the issue of the radiation dose loads that have to be expected due to the activated components/systems.

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Modal and response spectrum analyses of ITER divertor module

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While most of previous numerical analyses have been carried out under thermal and electromagnetic loads due to their significance, severe dynamic loads may also threat its structural integrity. The present study is to investigate resistance of complex ITER divertor module against typical seismic loads. Two kinds of huge finite element models, which consists of cassette body, inner and outer vertical targets, dome and stabilizers, were developed; one is simplified model without coolant tubes and the other is detailed model with three layered coolant tubes. At first, modal analyses to predict dynamic characteristics such as frequencies and mode shapes were conducted by employing either block-Lanczos algorithm or symmetric coupling algorithm considering water in coolant tubes. Subsequently, response spectrum analyses were performed with complete quadratic combination technique by taking into account different seismic magnitudes based on ASME (American Society of Mechanical Engineers) B&PV Sec. III Appendix N. As results, calculated stress intensities at critical locations were compared with corresponding design stress intensities, according to ASME code rule, of which details dependent on sensitivity parameters were discussed.

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Design of High Current Busbar Contact Connection for ITER Poloidal Field Converter

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The DC equipment of ITER poloidal field converter will be interconnected by DC busbar with water cooled aluminum busbars with cross section 200×60 mm, whose segments will be connected by aluminum flexible links in order to compensate that of thermal expansion. Because the contact surface between DC busbar and flexible links are small and the high current up to 30 kA flowed through, the power dissipated in the contact surface are about 900W, if the contact resistance is assumed only to be 1 uΩ, it is necessary to decrease the contact resistance in order to reduce the temperature rise of contact joint. This paper presents the different design on high current bolted busbar connection to increase the contact area and reduce contact resistance. Firstly, several methods to be discussed to reduce contact resistance based on lots of references. Secondly, these methods are analyzed by using ANSYS software, the pressure and stress between the contact surface is provided, it is obvious that the contact resistance is smaller with the increase of pressure and stress between contact surface. According to comparison among the different design, the best one is decided to be applied, the contact resistance is less than 1 uΩ. Finally, the experiment data are present to prove the validity of the design, the average temperature rise of connect terminal is less than 20℃, it can meet the requirement of ITER operation.

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Progress on thermo-hydraulic and thermo-mechanical performances of Helium-Cooled-Molten-Lead-Ceramic-Breeder as near-term alternative blanket for EU-DEMO

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Within the framework of EUROfusion activities, an alternative Helium-Cooled Molten Lead Ceramic Breeder (HC-MLCB) solid breeding blanket is being also developed at KIT for European DEMO. This concept is proposed as an alternative near-term breeding blanket and it is based on a fission-like "fuel-breeder pin" assembly configuration. Molten lead is used here as the neutron multiplier, Li4SiO4 in form of pebbles inside the fuel-breeder pins as tritium breeder and pressurized helium as coolant. In comparison to typical former cooling-plate configurations, the fuel-breeder pin assemblies greatly reduce the pressure drop, solving the key technology readiness issue of the currently available helium-circulator. Also, the combination of lead and lithium ceramics shows a good tritium breeding performance in a compact configuration (outboard blanket average radial thickness of 1000 mm instead of former 1300 mm).
After initial design works with this concept for the previous EU DEMO tokamak baseline design 2015 (reported elsewhere), the current status of the design activities on the HC-MLCB integrated for the latest EU DEMO baseline 2017 (EU DEMO 2017) are presented and discussed in this paper.
Firstly, the structural integrity of the preliminary blanket design under an in-box LOCA event (a key design driver) has been assessed and the design iterated to fulfill the design criteria. Then, and after corresponding neutronics analysis to validate the soundness of the design in terms of nuclear performance, thermo-hydraulic analyses have been conducted to evaluate the blanket temperatures and the coolant pressure drop. After some design iterations to satisfy the temperature design limits and pressure drop requirements, a structural assessment under normal conditions has been conducted with respect to the structural design standard RCC-MRx. The results presented here show that the current design of the HC-MLCB meets the basic nuclear and thermo-hydraulic-mechanical performances, setting the path for a consolidated design of this concept.

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10 MW FULGOR power supply performance tests and overview of test facility components

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The 10 MW FULGOR test facility (Fusion Long Pulse Gyrotron Laboratory) is built to meet the future needs of the gyrotron development, in particular for future fusion machines like the DEMOnstration power plant (DEMO). In the final stage it will enable KIT to test gyrotrons up to 4 MW RF output power in CW and frequencies up to 240 GHz. One of the main components of FULGOR test facility is the High Voltage DC Power Supply (HVDCPS), which has been developed and provided by Ampegon AG. The 90 kV/120 A CW PS is a so-called Enhanced Pulse Step Modulator which allows tests on gyrotrons with multi-staged depressed collectors. Additionally a Pulsed Power Supply (PPS) extends the capabilities of the HVDCPS to 130 kV/120 A for short pulses up to 5 ms every 2 s.

First tests with the HVDCPS 90 kV/120 A have been performed successfully on a 750 Ω test load. The pulse length in this phase was limited to 666 ms and maximum duty cycle of 0.22 % due to the limitation in the test load. The paper describes the tests that have been performed to determine the capabilities of the HVDCPS, like short circuit test to determine the power dissipated into an arc (<10 J), maximum rise time (10 % - 90 %) <50 µs, output voltage ripple at 90 kV (<0.33 %), settling time (<100 µs) and modulation up to 5 kHz.

Additionally an overview of the key components of FULGOR test facility, such as the PPS, Body Power Supply (BPS), 10 MW water cooling system, control and data acquisition system and the superconducting magnet with a magnetic field up to 10.5 T are briefly summarized.

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Verification of hydraulic performance for the DEMO divertor target cooling

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This paper presents the design activities and test of a vertical target mock-up, developed under the pre-conceptual design phase for DEMO Work Package DIV-1 "Divertor Cassette Design and Integration" under EUROfusion Power Plant Physics & Technology (PPPT) program.
The activities about the Divertor Outboard Vertical Target cooling mock-up are presented in term of CAD model (CATIA), thermo-hydraulic numerical simulation (ANSYS-CFX), structural analysis (ANSYS Mechanical), structural integrity verification (RCC-MRx) and manufacturing procedure. Moreover, the mechanical dimensions of support systems for plasma facing components (PFCs), manifold and diffuser have been analysed in detail, in order to avoid structural fault during test.
Test procedures are discussed, taking into account design parameters, design code and facility performances. The actual alloy (CuCrZr) selected for PFCs of EU DEMO divertor has been used also for the mock-up ,while two options are still under evaluation for manifolds/diffuser, CuCrZr and stainless Steel 316L(N)-IG, depending on the welding technology. Since the mock-up is mainly intended to verify hydraulic performances, it has been simplified by removing the W monoblocks from its PFCs.

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Supporting analysis of the ITER TBM Frame and Dummy TBM designs

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The validation and testing of tritium breeding blankets concepts, which are relevant for a future commercial reactor, is one of the goals of the ITER project. To achieve these objectives, mock-ups of breeding blankets, called Test Blanket Modules (TBMs), are tested in three ITER equatorial ports. Each TBM and its associated shield form a TBM-set that is mechanically attached to a steel frame. A frame and two TBM-sets form a TBM port plug (TBM PP). In case a TBM-Set is not available, it can be replaced by a Dummy TBM, steel made only.
Both the design and manufacture of TBM Frames and Dummy TBMs are fully under the responsibility of the ITER organization.
This paper describes the summary of recent analysis activities to support the design of the TBM Frame and Dummy TBM, which is presently in the preliminary design phase. In the following, the main items addressed in this work are reported.
• A refined evaluation through full 3D models of the heat transfer coefficient (HTC) and pressure drop of water cooling system of TBM Frame and Dummy TBM; alternative cooling circuits studied, with the aim of minimizing thermal stress and pressure drop.
• Electro-magnetic (EM) analyses on the most recent design of TBM-sets and a TBM PP with two Dummy TBMs under cat. II, III and IV disruptions;
• Evaluation of the impact of the changes from CDR configuration on EM loads on Dummy TBMs
• Detailed thermo-mechanical analyses and structural integrity assessments of TBM Frame and Dummy TBM carried out according to RCC-MR 2007.
• Proposal of design improvement from the outcomes of the above analyses, to be possibly included in the FDR configuration.

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

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On the path towards a Metal Foil Pump – Latest results and new experimental facility

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The current design baseline for the EU DEMO implements the KALPUREX process for the fusion fuel cycle. This process aims to reduce the tritium inventory by separating hydrogen from other gases within the tokamak building and feeding it back to the matter injection system. The best candidate for the hydrogen separation unit close to the torus is a metal foil pump that relies on the effect of superpermeation. For a comprehensive investigation the dedicated HERMES setup at KIT was successfully exploited during the past years. This setup was used recently to demonstrate remarkable improvements in performance. However, due to the limits of operation of this setup a complete redesign was indispensable. Consequently, an improved facility was designed and assembled to overcome the previous limits. This HERMES plus facility allows an extended pressure range for operation due to a different plasma source as well as an advanced metal foil module. This modification aims to fill the missing gap between the operation pressure requirements for use in DEMO and the capability of previous setups at KIT and in literature.
In this paper two aspects will be presented. On one hand the latest results of superpermeation, achieved with the previous setup, will be shown, highlighting the progress in performance and understanding. On the other hand the new facility will be introduced, demonstrating the motivation of this evolution, the current commissioning status and finally the expected outcome.

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European DEMO divertor R&D activities: loads, design concepts and technologies

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As the first funding period of EUROfusion Consortium is nearing its end, the work package "DEMO divertor" (WPDIV) is entering the final project period concluding its preconceptual design activities. The primary mission of WPDIV is to deliver holistic design solutions of the divertor targets as well as the cassette and to assure the availability of required technologies at least with a preconceptual maturity. Numerous tasks have been performed in WPDIV covering the full spectrum of design study and technology development including design rationales, CAD models, multiphysics analysis and load specifications, cooling scheme, design optimization, joining technology, mock-up fabrication, inspection, high-heat-flux (HHF) fatigue tests, post-examination, failure modelling, and guidelines for structural design and analysis. After 5 years of the preconceptual phase, a remarkable progress has been achieved with successful outcomes. In this contribution, an overview and the latest results of WPDIV works are presented. One of the highlights will be the recent HHF qualification testing campaign where several different design concepts of plasma-facing components were evaluated in terms of fabrication quality, HHF fatigue performance and structural reliability using small scale mock-ups. All design concepts shall be subject to down-selection process to be made eligible for the subsequent industrial upscaling stage. Further highlights are specification of loads and optimization of the cooling circuits for the entire divertor cassette system. To this end, an integrated multiphysics approach was employed covering neutronic, thermal, hydraulic, electromagnetic and structural stress analysis. In addition, a brief overview is given on the supporting tasks such as corrosion (coating, testing), inelastic design guidelines, CAD of cassette fixation system and non-destructive inspection techniques. Finally, the future R&D strategy is briefly outlined, including an indicative description of the planning and preparation for the following conceptual design phase.

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Preliminary investigation on W foams as protection strategy for advanced FW PFCs

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Among the eight core missions towards the realization of nuclear fusion, a future reactor must ensure efficient and safe power exhaust through the divertor and First Wall (FW). The greatest challenges arise from the occurrence of plasma transients. A simulation of a DEMO-like FW Plasma Facing Component (PFC) was carried out assuming Vertical Displacement Event (VDE) and ramp-up limiter conditions. The results highlighted an extreme heat flux impinging the thin tungsten (W) armour produced by Plasma Vapour Deposition (PVD). As a consequence, localized surface vaporization, melting and re-solidification may occur. Vapour shielding and surface melting play an important role in terms of heat flux reduction, but the failure of the component may occur owing to the high thermal stress and cracking developed, which can compromise safety and prompt return to normal operation. A possible protection strategy is to provide the plasma a sacrificial structure able to promote the flux reduction while preventing the substrate from the failure. W-based refractory foams are a viable solution owing to their peculiar mechanical and thermal management capabilities, that can be tailored by choosing an optimum value of relative density and porosity. However, a proper FE modelling is challenging due to their complexity and anisotropy. Nevertheless, a recent study confirmed that the mechanical behaviour of an existing open cell foam is achievable through the calibration of an equivalent FE model. The present investigation aims to verify the feasibility of W based open cell foams as a possible sacrificial material for FW protection strategy. Thermo-mechanical analyses have been carried on the equivalent FE model of foam exposed to heat loads expected in VDE and limiter conditions. As a consequence, optimal values of relative density and porosity of foam are proposed to achieve the required protection capabilities.

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Sensitivity of First Wall thermal-mechanical performance on cooling channel geometry and thermal conductivity

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The First Wall (FW) of DEMO or following fusion power reactors will be exposed to high heat fluxes by thermal radiation and energetic particles from the plasma. During steady state, values of over 1 MW/m² are expected for the EU DEMO concept. The function of the FW therefore relies on (1) good thermal conduction from the plasma facing surface through the channel material, and (2) good heat transfer from the channel wall surface into the coolant medium flow. Those aspects influence the shell-average operation temperature of the structural material - determining the materials mechanical strength - and also the temperature spreads within the component - causing thermally induced secondary stresses.

For given boundary conditions, FW designs can be thermal-mechanically optimized. In practice, deviations between the optimum design point versus the device under service have to be accepted due to manufacturing tolerances and variable coolant conditions for variably sized channels. This paper assesses sensitivities of the temperature and stress fields by a parameter study performed by finite element analyses, considering also feedback of the channel geometry to the relative flow ratio through individual channels, using validated correlations for helium thermal-hydraulics.

Further consideration is given to the materials thermal conductivity, which can vary with alloy composition and material history. Thermal conductivities of FW candidate reduced activating ferritic/martensitic steels are reviewed, and own measurements are reported. The possible tolerance in the thermal conductivity due to tolerances in the alloy composition is assessed by application of a neural network trained specifically to the named family of steels.

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Failure and Melting of Intentionally Misaligned Tungsten Castellated Blocks under High Heat Flux

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High heat load test were performed by using 1) E-beam for tungsten blocks and divertor mock-up, and 2) Long pulse H-mode plasmas in KSTAR for tungsten blocks mounted on stainless steel base.
Tungsten blocks are exposed to a heat flux of 13 MW/m2 from the top with a beam spot size around 11.5 mm in diameter, 100 kV and 12.5 mA, while the divertor mock-up is exposed to much higher heat flux up to 130 MW/m2. In the case of KSTAR experiments, inter-ELM heat flux at the central divertor is in a range between 0.5-3 MW/m2, while that during ELM is up to 50 MW/m2.
Two tungsten blocks exposed to 13 MW/m2 e-beam show that the Cu interlayer is melted within 4 sec, while the surfaces of the tungsten layer show recrystallization.
From COMSOL modeling, the temperature of Cu interlayer reaches its melting point (1100 deg C) in 3.7 sec, which is consistent with the experimental observation: The temperature of tungsten layer is around 2100 deg C well above the recrystallization temperature (1400 deg C). With the extreme case of 130 MW/m2 and longer exposure time, Cu interlayer is completely melted and some part of CuCrZr base is melted down. Tungsten blocks exposed to KSTAR H-mode shows melting of Cu interlayers with closed gaps between two tungsten blocks by molten Cu. Simulation with q=3 MW/m2 shows that an exposure time of 15.8 sec is required to elevate the temperature of Cu interlayer up to the melting point. These results indicate that bonding between tungsten and Cu layers will be failed at a temperature over 1000 deg C resulting in poor thermal contact. Melting of Cu interlayer also causes further misalignment, which can cause further increase of temperature of the tungsten layers up to melting point of tungsten.

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Activation foil measurements at JET in preparation for D-T plasma operation

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In the frame of the JET3 Deuterium-Tritium (D-T) technology project within the EUROfusion Consortium program, several neutronics experiments are in preparation for the future high performance D-T campaign at the Joint European Torus (JET). The experiments will be conducted with the purpose to validate the neutronics codes and tools used in ITER, thus reducing the related uncertainties and the associated risks in the machine operations. One of the requirements for the implementation of successful benchmarks is the accurate measurement of neutron fluence during the irradiation phase. The development of measurement techniques for the accurate monitoring of neutron fluence in the device as well as in the surrounding areas is a crucial aspect with respect to neutronics code validation for shielding design, radiation protection and safety assessment. Among others, the foil activation technique is used at JET to evaluate neutron streaming in regions far from the plasma source and along large shielding penetrations as well as to provide an estimate of the neutron spectra to complement shutdown dose rate measurements.
In the present study, the neutron fluence measurements performed using activation foils during the JET Deuterium-Deuterium (D-D) campaigns are discussed. The activation foil results are compared against other experimental techniques and Monte Carlo simulations and a satisfactory agreement is observed. Moreover, the pre-analysis of the experiments to be performed in the forthcoming JET Tritium-Tritium (T-T) and Deuterium-Tritium (D-T) campaigns is presented. The results of the study provide important data for the implementation of experimental activities at JET and support the preparation in view of the planned D-T operation, allowing the exploitation of the unique 14 MeV neutron yields anticipated. Furthermore, they contribute to the validation of the tools employed in nuclear analyses, which are fundamental for the design and safety of ITER and future fusion power plants.

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MIMO shape control at EAST tokamak: simulations and experiments

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Over the last few years, new magnetic control algorithms have been developed and tested on the EAST tokamak. The aim is to improve the overall plasma performances and to open the way to the control of advanced plasma magnetic configurations [1]. In order to achieve such an objective, an architecture based on a MIMO plasma shape controller was proposed in [2].
This architecture relies on specific control algorithms for plasma vertical stabilization and for the current control in the Poloidal Field (PF) circuits. These components have been designed exploiting the CREATE magnetic models. In particular, a voltage-driven Vertical Stabilization system, a controller for the plasma centroid position, and a multi-input-multi-output (MIMO) PF currents controller were implemented and experimentally validated in 2016-2017 [3] [4].
The plan for the 2018 EAST experimental campaign is to test and validate the MIMO plasma shape controller that relies on the architecture tested in 2016-2017, and that adopts an approach similar to the one used by the JET's eXtreme Shape Controller. Such an approach will enable the integrated control of the plasma boundary and of the heat flux on the divertor plates.
In this paper the simulation results obtained with the CREATE modelling tools are compared with the ones obtained experimentally for different setups of the magnetic control system.

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High resolution scanning electron microscope for sequential testing and analyses of full-size PFC components of AUG

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The complex power and particle wall loading conditions in fusion devices lead to various surface modifications of plasma-facing components (PFCs). To assess the consequences of these modifications on power handling capability and lifetime of PFCs, detailed microscopic studies of the surface and internal structure are required. Essential are analyses of the same area before and after plasma exposure. Non-destructive analyses are mandatory for such sequential testing, i.e., the complete tile as installed in the fusion device must fit into the microscope. Therefore, a scanning electron microscope (SEM) with focused ion beam (FIB) and analytics, energy and wavelength dispersive X-ray spectroscopy (EDS/WDS), was procured and commissioned at Max-Planck-Institut für Plasmaphysik.

This SEM is equipped with a newly developed heavy-duty stage, which allows to analyse samples up to a mass of 10kg, a length of 44cm, and a height of 10cm without and 6cm with additional rotation module. The accessible area on the sample is 23x10cm2. The achieved imaging resolution is better than 5nm. Cross-sections can be prepared by FIB. A multiple gas injection system enables, e.g., to coat markers for erosion measurements. Elemental mapping by X-ray spectroscopy is possible also on FIB cross-sections. Small features (tens of nanometers) can be investigated by using low electron beam energy (3-5keV).

In this contribution selected SEM analyses using FIB and EDX/WDX capabilities will be presented from material erosion and deposition experiments on divertor and first wall tiles exposed in ASDEX Upgrade (AUG). The examples include analyses of tiles with controlled pre-exposure damage structures and of W monoblock mock-ups pre-damaged by high heat flux testing after their exposure using the AUG divertor manipulator. Also data from heavy-alloy AUG divertor tiles installed for an entire campaign in AUG will be presented. The potential of the analyses capabilities of this SEM device will be elucidated.

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Influences of fabrication conditions on hydrogen isotope retention in W coatings

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Low pressure plasma spraying (LPS) and spark plasma sintering (SPS) are attractive techniques to prepare W armor layers on substrate materials. The properties of LPS-W and SPS-W depend on fabrication conditions. In this study, LPS-W and SPS-W layers were prepared on graphite and carbon fiber reinforced carbon composite (CFC) substrates at different temperatures, and D retention after plasma exposure was examined.

LPS-W layers were prepared on graphite tiles (IG-430U) at 1073–1353 K or 1233–1453 K. The coatings were subjected to heat treatment in vacuum or mechanical polishing to remove fine W oxide grains formed by fume condensation. SPS-W layers were prepared on CFC tiles (CX-2002U) at 1773, 1873 and 1973 K. The coating thickness was 1 mm. After removing the substrates by mechanical polishing, the coatings were exposed to D plasma at 373 K in a linear plasma device. The flux and fluence were 5×10^21 m^-2 s^-1 and 2×10^25 m^-2, respectively. The incident ion energy was 80 eV. Contents of D and H were measured using thermal desorption spectrometry at 0.5 K/s.

For all types of coatings, the D retention was in a range of 6–35×10^19 D m^-2. The main desorption peaks appeared at 500–700 K. Fine W oxide grains on LPS-W were successfully removed by heat treatment and polishing. The D retention in the polished samples was slightly higher than that in non-polished ones. The coatings prepared at 1073–1353 K showed smaller D retention than those formed at 1233–1453 K, though columnar grains were more developed in the latter. The D retention in SPS-W was insensitive to the preparation temperature and slightly smaller than that in LPS-W, though the concentration of impurity H was higher. The correlation of D retention with microstructure and surface morphology will be discussed.

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Possibility study of the partial neutron calibration for neutron flux monitors in torus devices

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The absolute calibration of the detection efficiency for the total neutron yield in the whole plasma is one of the most important issues in the neutron diagnostics such a neutron flux monitor (NFM). In many magnetic confinement devises, those neutron detectors are calibrated by moving or rotating a neutron source such as a Cf-252 radioactive source or a compact neutron generator on the magnetic axis in the vacuum vessel. The in-situ calibration work needs a long time and a big effort. Therefore, we consider the possibility of the partial neutron calibration, where the neutron source is located in the vacuum vessel only near the detector or limited number of the toroidal positions, from the in-situ neutron calibration data of the Large Helical Devise (LHD), JT-60U and the other tokamaks. The partial calibration is very useful to estimate the detection efficiency prior to the full calibration, especially on ITER, and might be a final calibration instead of the full calibration. In the case of the NFM using a U-235 fission chamber located outside of the vacuum vessel on the equatorial plane, larger than 90% of the detector counts are contributed by neutrons in the toroidal angle range of -60°~ +60°. The detection efficiency for the neutron source located one toroidal position on the magnetic axis (called point efficiency) decreased exponentially with the absolute toroidal angle of the neutron source. We found that a partial calibration in the toroidal angle range of -60°~ +60° combined with exponential extrapolations to -180° and +180°can estimate total detection efficiency within 5% uncertainty. In the case of the point efficiency measurement at the limited number of the

toroidal locations, typically 10-20 locations, the total detection efficiency for the whole plasma can be estimated within 5% uncertainty by the interpolation using the MCNP simulation result.

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Contribution to safety analyses of DEMO HCPB using AINA code

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The main motivation of the current work, framed under the safety EUROfusion activities to develop DEMO, is to present the conclusions drawn from our contribution to the safety studies of the HCPB DEMO design carried out by the team tasked with AINA code development. During 2016 and 2017 a new AINA version has been built and properly validated in order to evaluate plasma evolution and in-vessel components strains inside the European DEMO designs. As a result, AINA is able to simulate several accident scenarios as plasma disruptions or structural material meltings due to a LOPC (Loss Of Plasma Control) and in-vessel melt either of FW, blanket structure and/or divertor modules because of thermal stresses due to a LOCA (Loss Of Coolant Accident). After due analysis, it has concluded that it would be desirable to carry out a possible design review focused on ensuring a suitable operating temperature range with a bigger safety margin for all the materials which make up the HCPB BB, as well as the need to guarantee a quick detection and actuation by means of a proper system, depending on the affected equipment, when the most demanding transients take place which may drive the reactor to suffer a melting scenario and a very energetic plasma disruption at the same time. These events include an increase of fueling injection above 50%, a permanent improvement in the confinement time and a punctual impurity increase above 300%. Other perturbations has been studied, which some of them only provide information on non-dangerous cases as a decrease of the fueling injection rate, impossible situations from a technical or physical point of view as an unexpected and sudden increase of external power injection above 630% or melting processes as LOCAs (for any severity level) or a decrease of the external power injection.

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Investigation of divertor movement during disruptions

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The divertor, being the main power exhaust of a tokamak, is exposed to high heat fluxes and therefore must be precisely aligned to prevent leading edges. Since the transition from carbon to tungsten tiles in ASDEX Upgrade it was found that a specific assembly in the divertor was misaligned up to 1.5 mm after the experimental campaigns. This lead to heat spikes on the edges of several tiles and subsequent melting. To understand the origin of this movement, numeric model was created in ANSYS Maxwell containing the full 3D coil setup of ASDEX Upgrade, the plasma current and a segment of the divertor assembly. The plasma current (1.6 MA) was set to decay within 2.5 ms to 10 ms in order to create a poloidal field change of 50 T/s to 200 T/s while the toroidal field was constant at 2.4 T in the divertor region. The simulation revealed a parasitic current flowing from the support structure through the outer tiles. The resulting jxB force can reach up to 2500 N and the torque up to 780 Nm on a single tile depending on the poloidal field change. For carbon tiles, the maximal force and torque are five times smaller due to the higher electric resistivity. A current flowing through the conducting support of the assembly through the vessel wall

and back through the assembly causes additional force (max. 5500 N) and torque (max. 2300 Nm). The other support is isolated by SiN plates. A friction test showed a static friction coefficient of 0.1 under normal forces larger than 5 kN. A FEM model, using these results, showed that the friction force at the SiN plates is overcome and the assembly is moved.
The models and detailed results of the calculations will be presented at the conference.

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Deuterium retention behavior in tungsten irradiated with neutron under divertor operation temperature

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In the fusion reactor, tungsten will be exposed to high heat flux, neutrons, ash and fuel plasma of fusion reaction including tritium. The irradiation defects generated by neutrons will dynamically migrate, which results in the accumulation and annealing of irradiation defects. The irradiation defects in tungsten will act as potential trapping sites for hydrogen isotopes and, therefore, increase the hydrogen isotope retention.
Tungsten samples were irradiated by neutrons in HFIR (High Flux Isotope Reactor) in ORNL (Oak Ridge National Laboratory) up to 0.5dpa at temperatures of 1073 and 1373 K (named as AW-51 and AW-53 according to the sample ID in ORNL, respectively), which are equivalent to the divertor operation temperature in DEMO. Then, the samples were exposed to deuterium plasma at 673 K, and deuterium retention was evaluated by TDS (Thermal Desorption Spectroscopy) conducted in INL (Idaho National Laboratory).
The deuterium desorption spectrum for AW-51 showed deuterium desorption peak at around 850 K. That of AW-53 was also at the same temperature. This indicates the species of irradiation defects should be the same between these tungsten samples. Besides, deuterium retention in AW-53 was almost half compared to that of AW-51. It was suggested that the irradiation defects induced in tungsten annealed during neutron irradiation under high temperature. Consequently, deuterium retention was reduced for AW-53 due to the lower concentration of irradiation defects.

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Scaling analysis and design for the test model of water-cooled ceramic breeder blanket

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In Chinese Fusion Engineering Test Rector (CFETR), blanket is a key component, responsible for producing and transporting tritium, energy conversion and output, so its safety is of particular concern. The water-cooled ceramic breeder blanket (WCCB) is one of three candidate blankets for CFETR. To confirm safety of WCCB, sufficient data are required to estimate the thermal-hydraulic state and response in the blanket during postulated accidents. Due to strict test conditions and the complexity of WCCB structure, the expense of direct experiments is so great that it is inevitable to design a simplified and scaled-down model instead of the prototype for experiments with some appropriate scaling techniques. In this paper, based on same working fluid—water, full-temperature and full-pressure test conditions, scaling analysis is used. On one hand, the top-down scaling analysis is adapted to preserve the system dynamic responses. On the other hand, the bottom-up scaling analysis is adapted to preserve effective local phenomena. Through that, the characteristic time ratio group and the relation of important parameter ratios are established. The design scheme of reduced height and the number of flow channels is adopted, and the prototype blanket's structure is reasonably simplified to reduce difficulties of manufacture. Finally, scaling distortions of designed model are quantified, and verify that model can satisfy engineering application requirements. The test section can competently reproduce the thermal-hydraulic behaviors in WCCB.

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Layered W-WC composites prepared by FAST

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One of the main difficulties of designing fusion reactor is the development of plasma-facing materials that have to be resilient to the proximity of plasma. Pure tungsten is a primary candidate for this material but has to be strengthened either with particles or fibers to improve its' brittleness at moderate temperatures and inhibit recrystallization as well as grain growth at higher ones.
To limit the W grain growth, we proposed the incorporation of tungsten carbide particles in tungsten matrix. Samples were prepared from W and WC powder mixtures and consolidated with field assisted sintering technique (FAST). Layered W-WC composites with the gradually increased content of carbide phase were prepared to tailor the thermal conductivity of the material for monoblock of divertor. The layering will reduce thermal shocks and control heat transfer to the copper-based cooling system.
We have already confirmed, that sintering of W and WC particles by FAST induces in-situ high-temperature reaction with the final composition of cubic W with hexagonal W2C phase. W-W2C composites exhibit high density and improved mechanical properties at room and elevated temperatures. If the W2C content in the composite is 5 wt % or higher, W-grain growth is inhibited even after aging at 1600℃ for 24 h, due to the pinning of W grain boundaries with smaller W2C. The mechanical properties of aged samples did not impair, and chemical composition remained unchanged. Layered composites with the gradually increased content of W2C was successfully prepared. If the difference in the carbon content between two layers is too high, the growth of elongated W2C grains is induced. XRD and EBSD analyses confirmed that these elongated W2C grains had preferred orientation (0,0,2). The microstructures of such phases were thoroughly examined.

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Cooling optimization of the electron cyclotron upper launcher blanket shield module

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The four ITER EC (Electron Cyclotron) Upper Launchers inject up to 8 MW microwave power each with the aim to counteract plasma instabilities during plasma operations. The structural system of these launcher antennas will be installed into four upper ports of the ITER vacuum vessel.

The structural part of the Upper Launchers which forms the plasma facing component is called the Blanket Shield Module (BSM) that during operation will be heated by nuclear heating from neutrons and photons, thermal radiation from the plasma and mm wave stray radiation. The BSM is classified as an ITER VQC1 component since together with its cooling system it is fully immersed into the torus vacuum environment.

Recently Fusion for Energy has started the manufacturing of a full scale prototype of the BSM with the objective of finding an optimum manufacturing route including both technical and economic aspects.

This paper describes the design changes that have been implemented for the optimization of the BSM cooling performance by adapting the design and manufacturing route according to the know-how of the prototype supplier (ATMOSTAT) of Hot Isostatic Pressure (HIP) technologies. In particular, a highly efficient flange cooling scheme has been implemented.

This paper also details the design process that has driven the cooling optimization, including finite element steady state thermal analysis and CFD (Computational Fluid Dynamic) analysis. Specifically CFD was used for balancing the flow of the parallel water channels.

Finally this paper illustrates experimental tests results of the qualification mockups used to validate the HIP procedures proposed by ATMOSTAT.

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The new attempts for the in-vessel pressure gauge in the KSTAR plasma

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The in-vessel pressure gauge refers to a vacuum gauge installed inside a vacuum vessel of tokamak. The inside of the vacuum vessel in which the fusion reaction occurs have to discharge the impurities and ashes those generated as a byproduct of the fusion reaction to sustain efficient state. Also the impurities and ashes of the plasma impinging on the divertor plate along the magnetic field are transformed into neutral gas and then exhausted. When the divertor contacting with the high-temperature plasma experiences a enormous heat load. And an in-vessel pressure gauge on the back of the diverter is required to analyze and control the amount of the particles impinging on the divertor.
In addition, the gauge calibration can greatly enhance the convenience of understanding the divertor design and analysis, and the physical phenomena of structures in vacuum vessels when expressed in terms of conventional vacuum pressure.
We have made new attempts to improve the usability and performance of the in-vessel pressure gauge. There are three new attempts: first, direct filament heating current setting; second, gauge calibration linked to gas input; and third, shot by shot calibration. In this paper we will describe detatils and pros and cons.

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Optimization of high heat flux components for DIII-D neutral beam upgrades

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Upgrade of the DIII-D neutral beams leads to enhanced heat loads on many components, such as pole shields, calorimeter and collimator. Higher power is now desired for the neutral beams, increasing from 2.6 MW to 3.2 MW per source leading to a normal heat flux loads of up to 55 MW/m2 for the calorimeter. Original designs experienced local melting and fatigue cracks during operation at 2.6 MW. The Princeton Plasma Physics Laboratory is responsible for the design and manufacturing of the upgrades of these components.
Heat flux distribution on neutral beam components is very uneven and leads to significant thermal stresses. High heat flux density impact requires surface optimization to reduce surface heat flux projection, and avoid localized melting. Several new design features were introduced to accommodate increased heat loads, such as molybdenum inserts for the pole shields, two-dimensional shaping for the calorimeter, and three-dimensional shape optimization and replaceable copper inserts for the collimator. Additionally, all three components include an optimized cooling system design featuring peripheral cooling of copper components. This takes advantage of the metal's high thermal conductivity to achieve cool down between the pulses, and, simultaneously, avoids high thermal stress situations where cooling channels are located close to the high heat affected zones.
The optimization process included applying analytical relations for the transient temperature distributions on the high heat flux components. These relations were confirmed by previous DIII-D experimental results. To validate the designs, numerical simulations were performed using ANSYS software and consisted of two stages: transient fluid flow simulation in conjunction with heat transfer analysis in the solid parts; and structural analysis using the temperature distribution obtained at the first stage. Special functions were introduced to include complex heat flux effects on the 3D surfaces. Results of the design optimization and numerical simulations will be presented.

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Interference fit process development for the ITER vacuum vessel gravity support mock-up fabrication

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The ITER Vaccum Vessel (VV) is supported by the nine VV gravity supports (VVGS) located on the cryostat toroidal pedestal. The VVGS is dual hinge type that fastened by dowel on the hinge-block hole. The primary hinge restrains a vertical and toroidal movement of the VV system against fast displacements by the seismic events or fast transients. The secondary hinge restrains steady vertical load. However, the hinges allow radial thermal expansion during temperature increase for operation (100°C) and baking (200°C). This paper presents the technical approach and result of interference fitting process of the sleeves and MoS2 coated dowel to the full-scaled VVGS mock-up. Since the sleeve and hinge-block hole have tens of micrometers tolerance and around two meter long length, shrink fit method has been selected for the interference fitting of sleeves. To secure a sufficient time for process, liquid nitrogen was charged to the handling fixture capped sleeve hole's cavity. As a result, the required contraction time was secured as hundreds of seconds. Since the moisture which could be released from the VVGS in the vacuum evironment might be affect the operation of the VV or cryostat, defrosting treatment of sleeve is required. A local protection system was selected with charged nitrogen injection nozzles. As a result, the defrosted surface of shrinked sleeve was maintained during fitting process. A selected MoS2 coating solution was confirmed to satisfy the lubricating ability and durability of technical specification through the pin-on-disk test. The central

heating bar was used to insure uniform thermal expansion of the sleeve hole. As a result, MoS2 coated dowel successfully inserted into the sleeve hole without surface contact. Consequently, the interference fitting was successfully done for VVGS mock-up fabrication, and the technical solutions will be applied to the VVGS manufacturing.

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Methods and strategies on thermal integrity management of the ITER Thermal Shield

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Due to its main function as provider of a thermal radiation opaque barrier to the superconducting magnets, the ITER Thermal Shield (TS from now) design guarantees an appropriate thermal behaviour during operation. All the methods and strategies implemented with this purpose on the design, manufacturing and assembly of the TS, constitute the so called TS Thermal Integrity Management. The scope of this paper is to provide a comprehensiveness description of the TS Thermal Integrity Management keeping focussed on three main aspects: last methodology for an accurate estimation of the TS heat loads; advanced thermo-hydraulic modelling; combined analytical-experimental method for sizing pipe orifices. The TS heat loads estimation has to be accurate in order to know the margins regarding the agreed limits with the Cryoplant and Magnets. The proposed methodology includes detailed thermal radiation view factors involving all the systems interfacing with the thermal shield. This methodology is applied on a simplified case and compared with previous methods, resulting on a better understanding of the level of conservatism and the accuracy of the estimated heat inputs on the TS. A detailed analysis of the coolant temperature rise on the cooling tubes is also described in combination with experimental data and fluid-dynamic analysis of the required pipe orifices. As a result, a pipe orifices sizing method is described and new thermo-hydraulic parameters are found for feeding the planned final flow balance analysis of the thermal shield. As a conclusion of the paper, it is presented how the described methods are integrated onto the strategy that aims to manage the actual component emissivity data, being a strong tool for assessing the impact of the silver coating quality (during manufacturing and assembly) on the expected TS thermal behaviour. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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Reactivity and thermal stability of ternary Be-Zr-V beryllides

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As a water-cooled solid breeder blanket of a fusion reactor, safety concern has become one of the most critical issues. In specific, Be pebbles as a multiplier have been well-known to generate hydrogen and exothermally react while a loss of coolant accident (LOCA) occurred. In contrary to these Be pebbles, Beryllium intermetallic compounds (beryllides) are one of promising materials because of its much more stable chemical reactivity at high temperatures. Currently, many works on the development of advanced neutron multipliers by Japan and the EU is part of the DEMO R&D activities at the International Fusion Energy Research Center (IFERC) project, which forms a part of the Broader Approach (BA) program. Fabrication methods of beryllides pebbles have been successfully developed by combining a plasma sintering synthesis method and a rotating electrode granulation method.

By using these methods, preliminary synthesis of the ternary beryllide pebbles with mixtures of Be13Zr and Be12V with ratios of 1:0.1, 1:0.4, 1:0.6, 1:0.8, and 1:1, has been conducted. It was clear that small size of Be13Zr phase as precipitate and Be12V phase formed without Be phase formation since those composition does not contain the peritectic reaction.
Additionally, the ternary beryllide pebbles found out to have a lower reactivity to water vapor as well as a higher thermal stability. In the present study, not only synthesis process but also characterizations at high temperatures will be introduced.

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Neutronic assessments towards a comprehensive design of DEMO with DCLL Breeding Blanket

On the way towards a comprehensive design of DEMO, step by step all the systems and components must be introduced as their definition or refinement progresses, in order to demonstrate the viability of a design on larger scale, i.e. leaving fewer margins to undetermined questions.
Among the EUROfusion Programme, new aspects have been recently fixed or furtherly developed as the Divertor, the First Wall (FW) and the Flow Channel Inserts (FCI) designs. Furthermore, the integration of Heating and Current Drive (H&CD) systems, as the Neutral Beam Injector (NBI), has started.
The introduction or modification of these systems and components could seriously jeopardize the nuclear behaviour of an initially validated Breeding Blanket (BB) DEMO concept, since many neutronics criteria - among others - could be no more fulfilled. Since the design of DEMO is a continuous upgrade under iterative process, as the refinement push on, most of the studies have to be repeated to demonstrate that criteria are still respected in a fully integrated design.
In this work the influence on Tritium Breeding Ratio (TBR) of a new design of detached FW protecting BB from extremely high heat fluxes is investigated. The impact of different typologies of FCIs is assessed also according to the degree of detail in the neutronic description. The divertor composition and its topology also reveal to have strong impact on responses apparently not related with its design, as the tritium production in the BB. Besides, the integration of NBI minimizing its invasiveness in the BB is verified by neutronic analyses concerning the main BB functions: fuel breeding and heat recovery. Accordingly, TBR and Nuclear Heating (NH) are assessed.
The study is performed for a Dual Coolant Lead Lithium (DCLL) BB DEMO although can be extrapolated to other BB concepts.

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Fabrication and characterization of Be12V pebbles with different diameters

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Neutron multipliers with lower swelling and higher stability at elevated temperatures are desired for the pebble bed blankets of designed DEMO rector. Among beryllium-based intermetallic alloys, vanadium beryllide Be12V is considered to be an attractive material from the point of view of its potential use as an advanced neutron multiplier of the breeding blanket. Preliminary assessment of its properties showed that, compared to other beryllides, Be12V has a small value of deuterium trapping efficiency and low rate of hydrogen generation by reacting with water vapor.
Rotating electrode method (REM) was applied in this study for fabrication of sphere-shaped Be12V pebbles using a plasma-sintered Be-V electrode. Be-V electrode as raw material played a role of the

target which was melted and centrifugally ejected in the form of spherical droplets in the helium-filled atmosphere of REM apparatus. Since the angular rotation speed of Be-V rod is one of the important technological parameters which significantly influences the sizes of produced pebbles, the wide operating range (2000-6000 rpm) was selected by granulation process and produced batches of pebbles were analyzed. The yield properties, sizes and some characteristics of microstructure of fabricated vanadium beryllide pebbles are summarized and discussed in this work.

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Design and development of the mechanical support structure for ITER in-vessel magnetic sensors

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A mechanical support structure (a.k.a. "platform") has been designed to provide mechanical support and thermal conductance for the inductive magnetic sensors installed on the inner shell of ITER vacuum vessel (VV) for equilibrium and high-frequency magnetic field measurement. The platform design is modular so as to simplify the on-site installation process. It consists of a permanent and a removable section. The permanent section is welded to the VV surface, forming the base of the whole assembly and allowing electrical connection with the in-vessel electrical service. The removable section houses the sensor head, which is engaged with the permanent base through a knife-switch type connector. Alignment of the two sections is realized via two alignment pins built into the permanent base. The removable section is designed to be compatible with remote handling operation in case of replacement. Real-scale prototypes of the platform components had been successfully manufactured and assembled together according to the design. Good alignment between the two sections has been achieved. Full electrical continuity from the sensor head down to the electrical service cabling has been demonstrated in the platform prototypes. Continued R&D efforts will focus on testing some installation and operation aspects, including mechanical vibration, thermal cycling, cable clamping, etc.

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The Design of the DONES Lithium Target System

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In the framework of the EU fusion roadmap implementing activities, an accelerator-based Li(d,n) neutron source called DONES (Demo-Oriented early NEutron Source) is being designed within the EUROfusion work-package WPENS as an essential irradiation facility for testing candidate materials for DEMO reactor and future fusion power plants. DONES will employ a high speed liquid lithium jet struck by a 125 mA, 40 MeV deuteron beam to generate the intense neutron flux used to irradiate the materials sample up to the desired level of displacement damage (~10 dpa/fpy for iron in 0.3 l) and He production rates (~10-13 appm He/dpa).
In order to rapidly achieve a sound and stable design, a new configuration of the DONES target system based on the so-called integral concept has been proposed as reference solution in place of the former baseline design that envisaged a target assembly endowed with a replaceable back-plate,

being the latter solution not fully qualified yet and thus not readily implementable.
Moreover, following the outcomes of a detailed dedicated analysis taking into account several integrated aspects (such as tritium generation, cavitation issues, maintenance strategy, etc…) a decision was taken to move the Quench Tank inside the Test Cell while in the former design it was arranged in the lithium loop area, below the Test Cell floor.
Further modifications have also been introduced concerning inlet pipe and vacuum chamber configuration as well as support structure layout. A more consolidated design of the interfaces with the lithium loop and the accelerator beam line has been proposed too.
In this paper, a brief description of the current design of the DONES target system is presented including all the above mentioned aspects. Main results of supporting neutronic and thermomechanical analyses are also reported, showing the capability of the system to fulfil the prescribed requirements.

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Plan and progress of the fusion neutron sources at KAERI for fusion and fission applications

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According to the National Fusion Energy Program in Korea, Volumetric Fusion Neutron Source (temporarily called, V-FNS) has been planned and Compact Fusion Neutron Source (temporarily called, C-FNS) development was started at KAERI, which can be used in the fusion and also the fission/industrial applications such as radiotracing isotope production, radiography, and so on, in which the various targets were considered in parallel. For developing the C-FNS, plasma generator based on RF ion source including RF driver/generator was developed and target has been designed, fabricated, and tested with the high heat flux test facility (KoHLT-EB) before assemblying with the ion beam for investigate its integrity under the heating condition. Test conditions were prepared with the preliminary analysis and compared with the test results. From this result, the design was optimized and optimized target design was proposed. So far, the C-FNS is successfully developed, and the preparation of the on-site C-FNS and further V-FNS design in detail is in progress.

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Magnetic interaction between a tokamak reactor and its reinforced-concrete building

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Magnetic interaction between a tokamak reactor and its iron reinforced-concrete basement has been studied using the analytical model and ANSYS electromagnetic code. When the magnetic material is used for tokamak building, the leakage magnetic field from the tokamak is enhanced due to the normal angle incidence of the magnetic field line to the magnetic material wall. As this study is motivated from the ITER building construction, we use the similar dimensional data on ITER building. In this study, the magnetic field is assumed to be stationary.
We have analyzed the influences of the magnetized material on the position of the magnetic field null in the plasma break down phase, and on the x-point location during the divertor operation. We assume the iron wall plate instead of discrete reinforced-concrete wall, so that this assumption provides the larger magnetic disturbance than that of the actual case with rebars. When we assume floor and bio-shield wall made of 1 m thick iron plate with the relative permeability of 100, the null regime is shifted to the inboard side in the vacuum chamber. This can be understood by image currents induced by the ceiling, floor and outer wall. As the magnetic null regime is in the small

magnetic field area, it is disturbed when it is away from the floor and the ceiling. Such shift could be adjusted by the PF coil current. On the other hand, the x-point location during divertor operation is not affected by the magnetic wall because of the dominant contribution of the large poloidal coil current.
While the effect of the magnetic material on the plasma performance seems to be not crucial, we further investigate the degaussing operation, the mechanical strength of the floor due to the magnetic force and the induction effect by PF coils.

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Optimization and adjustment of impact set-up for testing of insulated pads of ITER blanket module connectors and first wall

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Insulated pads are used on ITER blanket module connectors and the first wall; their main insulating function is to break any current loop between the shield block and vacuum vessel and/or between the first wall and shield block. The design of the pads consists of a cylindrical or prismatic body manufactured from NiAl-bronze, a ceramic insulating coating (Al2O3 or MgAl2O4) which is applied on interfacing surfaces of the pad body and on the shield block or first wall beam. The pads work in ultra-high vacuum conditions and at elevated temperature 100 – 300 oC. Static and dynamic loads up to 2 MN can act on the insulated pads during various plasma events in ITER.
The qualification program for the insulated pads consist of cyclic and impact tests. The tests are performed for two categories of loading anticipated from plasma events. The number of cycles, force and energy are different for each category and therefore the qualification takes into account the worst case conditions.
The impact tests are performed on a weight drop test bench, which has been designed and manufactured in JSC "NIKIET". A measurement system for the test bench consists of specially designed force sensors and accelerometers.
The description of the pad design and the impact test set-up, the results of analysis for justification of the set-up parameters, and the results of the test bench commissioning are presented in this paper.

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First thermal-hydraulic and thermal-mechanical analysis of a CO2-cooled solid breeding blanket for the EU-DEMO

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Helium Cooled Pebble Bed (HCPB) Breeding Blanket (BB) has been intensively studied for the EU DEMO. However, several feasibility issues remain for a HCPB-class DEMO reactor, namely the large diameter of the Primary Heat Transfer System pipework, the resulting large coolant inventory and large expansion volumes required after an ex-vessel loss of coolant accident, the limited operational experience with relevant size He-turbomachinery and the large circulating power, among other. Due to the larger density of CO2, the use of this gas as primary coolant for DEMO can lead to key advantageous features, mitigating most of the issues posed for He-cooling and resulting in a higher net efficiency than that of HCPB, as reported in a previous study. Therefore, a CO2-cooled Pebble Bed (CCPB) has been proposed as an alternative coolant to He for the EU-DEMO. After identifying that CO2 will have a negligible influence on the neutronic performance, making the CCPB's TBR almost equal to the HCPB's one (TBR ≈ 1.15), a full first set of thermo-hydraulic and thermo-mechanical analyses with the commercial code of ANSYS CFX are reported here. The analyses are based on the

newly proposed design of breeding zone (BZ) in the enhanced HCPB fuel-pin concept for the EU-DEMO. Such pin-type fuel elements have been already used in liquid metal fast reactors since the 1960s. The paper will show that, despite the lower heat transfer capability of CO2 with respect to He, the fuel-pin design breeding zone improves the thermo-hydraulic performance, meeting the materials' temperature requirements. For the thermal-mechanical analysis, the structural behavior under normal operation has been assessed according to the available codes and standards (RCC-MRx). The results show that the CCPB can satisfy the basic thermal and mechanical blanket requirements and that CO2 is a realistic option as primary coolant for gas-cooled fusion reactors.

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Exploration of a fast pathway to nuclear fusion: first thermomechanical considerations for the ARC reactor

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Progress in technological fields such as High Temperature Superconductors, Additive manufacturing, new diagnostics, and innovative materials, has led to new scenarios and to a second generation of Fusion Reactor designs. A new Affordable Robust Compact (ARC) Fusion Reactor, which meets its goal in a cheaper, smaller but even more powerful, faster way to achieve Fusion Energy, has been designed at MIT.

An investigation of the load-following concept is necessary, in order to prove its feasibility on ARC reactor. We started from ARC's most close to plasma component, the vessel: finite element analysis models have been designed and thermo-mechanical analysis have been conducted. Thermal fatigue remains the main issue. The study demonstrated that the vessel is able to survive some years in particular conditions such as high temperature and variable thermal loads. This is quite enough, since ARC's vacuum vessel is thought to be replaced no later than two years, due to neutron embrittlement and neutron induced activation.

Supports, divertors and connections between the two walls should be designed and investigated, to improve vessel's resistance to disruptions without causing hotspots and stress concentrations during thermal expansion. Simulations on supports and channels disposition demonstrated that setting channel's inlet section in the upper side of the vessel, near supports is the best choice.

Indeed, Inconel 718 is not a good candidate from radioactive waste point of view, due to its neutron-induced activation properties. Therefore, few other materials were investigated, knowing the main material properties needed for vacuum vessel's structure. Reference materials used in fusion experiments and projects mostly come up to be less attractive than Inconel 718; for example stainless steels 304L and 316L, which show poor thermomechanical properties. However, the study found out that Vanadium alloys such as V-15Cr-5Ti could be a good substitute of Inconel for ARC's vacuum vessel.

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Preliminary structural assessment of the HELIAS 5-B breeding blanket

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The European Roadmap to the realisation of fusion energy, carried out by the EUROfusion consortium, considers the stellarator concept as a possible long-term alternative to a tokamak fusion

power plant. To this purpose a pivotal issue is the design of a helical-axis advanced stellarator (HELIAS) machine equipped with a tritium breeding blanket (BB), considering the achievements and the design experience acquired in the pre-conceptual design phase of the tokamak DEMO BB. Therefore, within the framework of EUROfusion WPS2 R&D activity, a research campaign, aimed at the investigation of the structural behaviour of the HELIAS 5-B BB, has been launched at KIT in cooperation with University of Palermo. The scope of the research has been the determination of a preliminary BB segmentation scheme able to ensure, under the assumed loading conditions, that no overlapping may occur among the blanket regions. To this purpose, the Helium-Cooled Pebble Bed (HCPB) and the Water-Cooled Lithium Lead (WCLL) BB concepts, presently considered for the DEMO tokamak fusion reactor, have been taken into account.
A 3D CAD model of a HELIAS 5-B torus sector has been adopted, focussing attention on its far end regions, namely the triangular and bean shape regions. Due to the early stage of the HELIAS 5-B BB R&D activities, the considered CAD model includes homogenized blanket modules without internal details. Hence, in order to simulate the features of the HCPB and WCLL BB concepts, equivalent material properties have been purposely calculated and assumed. Moreover, proper nominal steady state loading scenarios, based on the DEMO HCPB and WCLL thermomechanical analyses, have been taken into account.
A theoretical-numerical approach, based on the Finite Element Method (FEM), has been followed and the qualified ANSYS v. 18.0 commercial FEM code has been adopted. The obtained results are herewith presented and critically discussed.

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Examination of ITER Central Solenoid prototype joints

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The ITER Magnet System will be the largest and most challenging integrated superconducting magnet system ever built. For the Central Solenoid (CS), cable – in – conduit - conductors (CICCs) of nearly one kilometre length are produced, but still, it will be necessary to connect several lengths together to wind the gigantic 110 tonnes coils. The creation of these superconducting joints is one of the most delicate parts of the assembly. There are three types of ITER CS joints: sintered joints, coaxial joints and twin – box joints. US ITER, the ITER Domestic Agency of the USA produced a prototype containing all three types of joints. The goal is to test the performance of the joints in the SULTAN facility of the Swiss Plasma Center, capable of reproducing close – to – service conditions: high magnetic field (up to 11 T background field), high current (up to 100 kA) and high mass flow rate of supercritical helium for cooling. The paper describes the results of a comprehensive examination campaign aimed at understanding the relation between fabrication and performance of the CS prototype joints. The test campaign combines advanced image analysis for the assessment of the void fraction of the conductors and dimensional measurements performed at different cross-sections of the joints, scanning electron microscope (SEM) and energy dispersive x-ray spectroscopy (EDX) to evaluate the quality of the contact between the strands and the sleeve / sole, as well as a full assessment of the welds according to the most stringent acceptance levels of the standards in force. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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Pressure tests supporting the qualification of the ITER EC H&CD upper launcher diamond window

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The Electron Cyclotron diamond window which is located inside the port cell serves, together with an isolation valve, as primary vacuum boundary between the ITER vacuum vessel, the transmission lines and the atmospheric environment and it functions as confinement barrier. The window consists of an ultra-low loss Chemical Vapor Deposition (CVD) diamond disk brazed into a metallic housing and it has to guarantee the compliance with very stringent nuclear safety requirements and an adequate transmission capability for high power mm-waves (1.31 MW at 170 GHz). The design of the window unit is approaching its final phase including design validation analyses, the development of a dedicated qualification program and prototyping activities.
In preparation of the testing of complete window prototypes, pressure tests were performed on a mock-up formed by a diamond disk (D = 80 mm, d = 1.11 mm) brazed to two copper cuffs. The scope of the experiments was to show both the capability of the joining between the disk and the cuffs to keep the required vacuum tightness as well as the integrity of the disk when exposed to pressure loads. Based on the requirements defined for the pressure scenarios during normal operation and off-normal events, a test program was developed accounting for cyclic tests at low pressure differentials and overpressure tests up to 2 bar pressure difference over the disk to simulate severe accident conditions.
This paper gives an outline of the ongoing development of the overall qualification program of the ITER torus window and reports specifically on the experimental set-up and the successful outcome of the pressure tests of the brazed diamond disk mock-up. The test program for the complete window prototype needs discussion and approval by F4E and ITER and will directly determine the final qualification program for the diamond windows during series production.

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Equilibrium evaluation for OP1.2a Wendelstein 7-X experiment programs

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Wendelstein 7-X (W7-X) is a modular advanced stellarator, which successfully went into operation in December 2015 at the Max-Planck-Institut für Plasmaphysik in Greifswald, Germany and continued to thrive at the experimental campaign OP1.2a (August-December 2017). The term modular stellarator refers to a generalized stellarator configuration with nested magnetic surfaces created by a system of toroidally discrete coils, providing both toroidal and poloidal field components, and designed with the aim to create optimum equilibrium properties. The optimization criteria included the high quality of vacuum magnetic surfaces, good finite beta equilibrium and MHD-stability properties as well as a substantial reduction of the neoclassical transport and bootstrap current in comparison to classical stellarators. The mission of the W7-X project is to demonstrate the reactor potential of the optimized stellarator line.

Equilibrium calculations are the basis for the mapping of various diagnostics in different pol

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Practical Implementation within the Electron Cyclotron Upper Launcher of the French INB Order of 2012

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The ITER project is being undertaken at Cadarache, France, to construct and operate an experimental nuclear fusion facility. The aim of this paper is the description of the implementation of the French Order of February 7, 2012, concerning Basic Nuclear Installation (also called "INB") within the European Union Domestic Agency (EU-DA), specifically on the Electron Cyclotron Upper Launcher (EC UL).

The EC UL will be used for the control and heating of the ITER plasma. The launcher includes in-vessel items (nuclear shielding, blanket module, port plug structure and optics) and ex-vessel first confinement items (diamond window, isolation valve, waveguides, miterbends, tapers and port plug back end).

According to the general rules of the French order, ITER Organization (IO), being the nuclear operator, has to monitor all activities related to the Design, Construction, Operation, Maintenance, Final shutdown and Dismantling of Nuclear facilities during their full life cycle at ITER. The EU-DA, as a tier 1 supplier of IO, applies a Requirements Management and Verification (RMV) process in order to track, control and verify all technical requirements applicable to EC UL components. EU-DA has duties regarding the compliance with the requirements propagated in his full supply chain performing Protection Important Activities (PIA). The methodology of this process will be illustrated for different EC UL Protection Important Components (PIC). The paper will go in more detail on the Nuclear Pressurized Equipment classification (ESPN) of the EC UL and its cooling circuits. The nuclear safety demonstration, traceability, validation of methods, qualifications, prototype tests, calculations and modelling, will be described with specific examples of EC UL.

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Feasibility studies of DEMO potential waste recycling by proven existing industrial-scale processes

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This paper is focused on the analysis of existing industrial-scale process for recycling of DEMO steel components (Eurofer, AISI 316L) and Lithium orthosilicates breeder. The aim is the assessment of their practical feasibility and the individuation of preparatory activities to be performed for facilitating and improving the recycling.
In detail, the thermodynamic analysis of recovering 14C from Eurofer and AISI 316L steels by decarburization processes was performed, based on practices used in steelmaking. The mass of species in the formed phases: gas, oxide, metal, were quantified. The effect of decarburization process on other critical elements contained in the steels like Cr, Mn, W, Ni and Mo was investigated as well.
The decarburization of steel is a process, currently used in steelmaking for producing Ultra Low

Carbon (ULC) Steel and High Chromium Steel, based on the reaction between Carbon and Oxygen, both dissolved in liquid steel, by forming CO gas that is removed by operating under vacuum and/or by inert gas like Argon or Nitrogen. Another process analyzed is the recovery of the expensive 6Li, from used Li4SiO4 solid breeder pebbles, in order to reuse it in the manufacture of fresh solid breeder and to evaluate the way of removing detrimental radioactive impurities. The most promising candidate process is the melting of pebbles with addition of new 6Li enriched material.
A general trend in the modern steelmaking industry is toward a higher degree of robotization and remote control; this evolution is in line with the need of processing of potential radioactive wastes taking into account the issue of their decay heat and dose rate. The study performed aims to identify and select existing and emerging steel melting technologies that might be suitable for recycling of activated steels and breeder materials of DEMO.

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Thermo-hydraulic analyses and fatigue verification of the Electrostatic Residual Ion Dump for the ITER HNB

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In the ITER Heating Neutral Beam Injector (HNB) the remaining charged particles after the neutralization process will be removed by an Electrostatic Residual Ion Dump (ERID) where electrostatic fields are used to deflect the ions that are so dumped on to five panels, each one composed of 18 separate CuCrZr Beam Stopping Elements (BSEs).
The thermal loads applied on panels were calculated for full beam-on power and for the most severe beam conditions (3 mrad beam core divergence and 30 mrad beam halo divergence) taking into account also possible off-normal conditions due to possible variation of the gas throughput in the Neutraliser channels that can affect the heat fluxes on the ERID.
A customized finite element (FE) code was developed to allow parametric and coupled solid-fluid simulations in the sub-cooled boiling conditions expected in the swirl flow of cooling channels. The model was used to verify the expected temperature at the inner cooling channels and at the external surface, the pressure drop, the Critical Heat Flux (CHF) and bulk boiling.
Thereafter, thermo-mechanical analyses and fatigue verifications of the ERID panels were undertaken with non-linear material properties by using elasto-plastic properties of CuCrZr and by considering the most demanding condition produced by breakdown and beam-on/off cycles during the operation. The thermal results produced by transient thermo-hydraulic analyses were applied as body load onto the mechanical model to calculate stress and strain fields. The analysis results were post-processed for the fatigue verification by evaluating local effects and creep-fatigue interaction.

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Leak detection design for ITER gas injection system

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The main functions of ITER Gas injection system(GIS) are providing gas fueling(H2,D2, T2, 4He/3He, N2/Ne, Ar) for plasma, wall conditioning operation and neutral beam injectors. If there is leak on the gas supply lines during ITER plasma operation state, abnormal gas composition will affect or have potential to affect operation. Furthermore, Out-leak of Hydrogen or Tritium from gas supply lines will lead to safety issues, therefore leak of gas supply lines especially H2(D2)/T2 lines shall be detected as soon as possible during the operation. Leak detection design for GIS H2(D2)/T2 gas

supply lines during ITER operation state is introduced in this paper.
Pressure monitoring of manifold is an effect way to check a problem of a line. Considering the handling of tritium, all the gas supply lines are enclosed in a secondary containment pipe(guard pipe), pressure of which shall be lower than the environment pressure. Since the nominal pressure of the gas supply pipe is also lower than the pressure in the interspace, by monitoring and comparing the pressure in gas supply pipe and guard pipe a leak of gas supply line could be identified with reasonable control logic.
Besides, hydrogen isotope may leak out to the interspace by diffusion. In this case Hydrogen detector and Tritium detector are employed to the interspace. The manifold interspace is continuously vented to Detritiation System with N2, therefore H2(D2)/T2 detectors are connected to monitor their concentration in the flow. Requirements that the interspace pressure are maintained to be lower than the environment pressure, at the same time the hydrogen isotope gas in the flow could be transferred to the detector as fast as possible are taken into consideration for the design of purge configuration and parameters(pipe diameter, flow rate, etc.).

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ECART analysis of the STARDUST dust resuspension tests with an obstacle presence

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The activated/toxic dust resuspension inside the vacuum vessel of future fusion devices as ITER or DEMO is a safety issue of main concern. In case of a LOVA or a LOCA, dusts produced during the normal and off-normal conditions can be released inside the tokamak building or towards the external environment. These accidents are not expected during the whole lifetime of the ITER machine, though in the past they were considered in the ITER Generic Site Safety Report with a conservative assumption for the lack of reliable resuspension models in the employed safety codes. To relax this strong assumption and validate resuspension models in fusion like conditions, different experimental campaigns in the STARDUST facility were performed at the ENEA laboratories in Frascati (Rome). In the first experimental campaign (2004), the resuspension of Tungsten (W), Carbon (C) and Stainless Steel (SS) dusts was investigated in an "empty tank" configuration, while the resuspension of the same dust types in presence of an internal obstacle was studied in the second campaign (2005). The obtained experimental results stressed that only a minor fraction of dusts is effectively resuspended. In the present work, focalized on the ECART code validation for the safety analysis of future fusion installations, a further step in the assessment of the semi-empirical "force balance" resuspension model implemented in the ECART code against the data obtained during the second experimental campaign (tank with an inner obstacle) is performed. The code predictions are quite in agreement with these STARDUST experimental data, and three charts (one for each dust type) were elaborated to predict the resuspension magnitude basing on the flow velocity on the structure where the dusts are initially collected.

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Experimental investigation on the interruption performance of a switch based on artificial current zero

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The quench protection switch (QPS) is indispensable to protect the magnet coils from the damage of a quench in a superconducting Tokamak. In this paper, a QPS based on the artificial current zero is involved. The vacuum circuit breaker (VCB), which is driven by a high-speed electromagnetic repulsion mechanism, is used as the main circuit breaker (MCB). Two kinds of commercial vacuum interrupters (VIs), which have electrodes generating axial magnetic field (AMF) and transverse magnetic field (TMF), respectively, are applied. Meanwhile, the breaking current with amplitude of 15-25kA is generated by a LC oscillating circuit. The countercurrent with frequency in range of 500~5000Hz is provided by a commutation branch. The interruption performance of the two VIs under different breaking current and frequencies of countercurrents is investigated. The experiment results indicate that the differences of interruption performance under the two types of magnet fields are enlarging with the increasing of the frequencies of countercurrents.

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Seismic analyses of the Double Closure Plate Sub-Plate for the ITER Electron Cyclotron Upper Launcher during the Vacuum Vessel baking scenario

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Four Electron Cyclotron Upper Launchers (EC UL) will be used at ITER to counteract magnetohydrodynamic plasma instabilities by aiming up to 20 MW of mm-wave power at 170 GHz. This mm-wave power will be injected through eight ex-vessel waveguide assemblies for each EC UL to the in-vessel waveguides. The power exiting the in-vessel waveguides located inside the Port Plug will be directed by quasi-optical mirrors to specific plasma locations. The Double Closure Plate Sub-Plate (DCPSP), which defines the border between ex-vessel and in-vessel components, was introduced in order to minimize the openings exposing the interior of the plug to avoid the near environment activation in case of maintenance or intervention on the in-vessel components. The seismic event taking place during the Vacuum Vessel (VV) baking scenario was identified as one of the most stringent load combinations for the DCPSP. The modal analysis of the DCPSP shows that the natural frequencies are far from the peaks of the ITER reference spectra at the Port Plug flange. Therefore, the feasibility of analysing the seismic event by using a static method as a replacement of the Response Spectrum method is also investigated. Then, the results due to the seismic event are to be combined with the ones produced due the loads occurring at the DCPSP during the VV baking scenario. The stress distribution produced from this load combination is categorized and compared with the allowable design limits in order to evaluate the mechanical integrity of the DCPSP. This work was supported in part by the Swiss National Science Foundation. This work was carried out within the framework of the ECHUL consortium, partially supported by the F4E grant F4E-GRT-615. The views and opinions expressed herein do not necessarily reflect those of the European Commission or the ITER Organization.

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Experimental and numerical studies on a gas flowing calorimetry for tritium accountability

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A pressure, volume, temperature, and concentration (PVT-c) method, which is widely used to measure the amount of tritium owing to its effectiveness, requires a desorption process from uranium

tritides and a transfer process to a measurement tank in the tritium storage and delivery system (SDS) for a tokamak-type nuclear fusion reactor. In addition, the repeated processes for the PVT-c, including the heating of the SDS bed, increase the potential risk of tritium release. For this reason, the application of other tritium measurement methods without the desorption and transfer processes is needed for the SDS bed. In this study, we designed and fabricated a uranium hydride bed with gas flowing calorimetry. Gas flowing calorimetry does not require desorption or transfer processes. Experiments on obtaining the calibration curve for the relationship between the amount of tritium and the difference in gas temperature in the calorimetry have been carried out. In addition, characteristics including the resolution, convergence, and reliability of the gas flowing calorimetry in a uranium hydride bed were obtained. Furthermore, the calorimetry was modeled and simulated under various steady state and transient conditions through a Multi-dimensional Analysis of Reactor Safety (MARS) code. The calorimetry loop consists largely of a uranium hydride bed, a circulation pump, flow controllers, chillers, heaters, buffer tanks, pressure gauges, thermocouples, and resistance temperature detectors. The uranium hydride bed contains 1893.75 grams of depleted uranium as a tritium storage material. A direct current power supply was used to simulate the decay heat of tritium.

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Thermo-Mechanical behaviour of ITER Blanket Modules interface between First Wall and Shield Block

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The Blanket System provides a physical boundary for the plasma transients and contributes to the thermal and nuclear shielding of the vacuum vessel (VV). It consists of modular shielding elements, the blanket modules (BM), which are attached to the VV. Each BM consists of two major components: a plasma-facing first wall panel (FW) and a shield block (SB). They are connected by means of a preloaded central bolt working in tension, and they further interface each other through a set of eight compression pads installed in each FW. The design of these pads has a major importance because of the very demanding Thermal and Electromagnetic (EM) loads they are subjected to. The thermal gradients taking place during the normal operation thermal cycling tend to bend the SB and the corresponding FW in different ways and they can induce an over loading of the pads or a loss of pretension with the risk of exceeding the damage load of the pads or of the bolt under the EM Major Disruptions and Vertical Displacement events. These phenomena have been numerically investigated on BM#14 (including a representative Enhanced Heat Flux FW) and BM#06 (including a representative Normal Heat Flux FW). A full set of hydraulic (CFD), thermal and structural finite element analyses and related structural (SDC-IC) assessments was performed, focusing on the behavior of the central bolt's pretension and on the reaction forces at the pads at significant instants of the thermal cycling. The results obtained demonstrated that the pads are within the allowable load for static conditions for all possible combination of thermal and EM loads. However the presence of a physical gap at some pads indicates that a fully conclusive study requires dynamic analysis techniques.

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Thermal-mechanical analysis and design optimisations of the IFMIF-DONES HFTM

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The International Fusion Materials Irradiation Facility – DEMO-Oriented Neutron Source (IFMIF-DONES) is planned to generate a high flux of 5E16 neutrons/s with First Wall relevant energy

spectrum. The High Flux Test Module (HFTM), is the dedicated assembly to bring the material specimens into the high flux region of the neutron source and maintain the specified irradiation conditions.

Based on neutronic analysis, the nuclear heating of the HFTM under normal operational conditions was calculated. A subsequent CFD analysis was performed to determine the temperature field of the HFTM structure. This temperature field and the internal operating pressure are the basis for the structural analysis and stress assessment of the HFTM body in regard of the requirements of the RCC-MRx code. Furthermore, design optimisations are presented especially in regard of the installation of the irradiation capsules in the HFTM during assembly. To keep installation time of HFTM in the Test Cell (TC) as short as possible, connectors are foreseen on top of the HFTM, which transmit signals and electrical power from and to the irradiation capsules. A design proposal of a quick and fail-safe HFTM-sided multi-coupling solution for these connectors is presented.

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Design and preliminar operation of a laser absorption diagnostic for the SPIDER RF source

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The ITER Heating Neutral Beam (HNB) injector is required to deliver 16.7 MW power into the plasma from a neutralised beam of H-/D- negative ions, produced by an ICP RF source and accelerated up to 1 MeV. To enhance the H-/D- production, the surface of the acceleration system grid facing the source (the plasma grid) will be coated with Cs because of its low work function. Cs will be routinely evaporated in the source by means of specific ovens. Monitoring the evaporation rate and the distribution of Cs inside the source is fundamental to get the desired performances on the ITER HNB. In order to proper design the source of the ITER HNB and to identify the best operation practices for it, the prototype RF negative ion source SPIDER has been developed and built in the Neutral Beam Test Facility at Consorzio RFX. In SPIDER, the dynamics of Cs will be monitored by measuring the emission intensity of the Cs 852 nm line along several lines of sight in the source. For a more quantitative estimation of Cs density a Laser Absorption Spectroscopy diagnostic will be installed; by using a wavelength tunable laser, the diagnostic will measure the absorption spectrum of the 852 nm line along 4 lines of sight, parallel to the plasma grid surface and close to it. From the absorption spectra the line-integrated density of Cs at ground state will be calculated. The paper will present the design of the diagnostic for SPIDER, with a description of the layout and of key components. The paper will also show the first experimental results from a preliminary installation of the diagnostic on the test stand for Cs ovens.

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Design and Countermeasures against Cavitation in a Downstream Conduit of the Liquid Lithium Target for International Fusion Materials Irradiation Facility

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A liquid-lithium (Li) free-surface stream flowing under a high vacuum serves as a Li target for the planned International Fusion Materials Irradiation Facility (IFMIF). As the primary Japanese activity for the Li target system of the IFMIF/EVEDA (i.e. Engineering Validation and Engineering Design Activities) project, implemented under the Broader Approach (BA) Agreement, cavitation-like acoustic noise was reported in the downstream conduit of Li target assembly (TA) of the IFMIF/EVEDA lithium test loop (ELTL), which aims to verify the lithium target and purification systems envisioned the IFMIF. To clarify the cause of this acoustic noise, we found that acoustic emissions due to cavitation occurred in a narrow area near the start of the bend pipe where the Li target impinged by using acoustic-emission sensors. And the method to determine this conflict (initial arrival) location was formulated. Intermittent high-frequency acoustic emission can also cause of the cavitation-erosion crack of the structural materials due to cavitation bubbles collapse (cavitation pitting). To examine the detailed location of the cavitation-like acoustic noise occurring in the downstream conduit of TA, the change of initial arrival position in the downstream conduit according to the velocity of Li target was calculated by the flow analysis of ANSYS/Fluent V16.2 and SCRYU/Tetra, and the design and countermeasures against this type of cavitation were discussed.

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Preliminary accident analysis of ex-vessel LOCA for the European DEMO HCPB blanket concept

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Based on the reference design HCPB2016 (helium cooled pebble bed) in the pre-conceptual design studies for the European DEMO, the primary heat transfer system (PHTS) for DEMO baseline 2015, and current parameter study for the plasma disruption conditions and the affected FW surface areas, ex-vessel LOCA (loss of coolant) with a double-ended guillotine break of a main pipe in the PHTS has been investigated that helium blows down into the tokamak cooling room (TCR). For the design basis accident (DBA) a fast plasma shutdown (FPSS) followed by a plasma disruption is assumed at 3 s after the detection of the LOCA at 80% of the nominal mass flow rate. Three main cases are identified: case I with the affected FW area of 0.1 m² in one loop and the mitigated plasma disruption, case II with 1.0 m² in two loops and the mitigated plasma disruption, and case III with 5.0 m² in two loops and the unmitigated plasma disruption. If EUROFER reaches 1000 ℃ an in-vessel LOCA takes place. Since this LOCA starts in case of the beyond design basis accident (BDBA) without the FPSS much earlier than it in the DBA, to save the computation time, the transport of source terms is performed for the BDBA. Also scenarios due to the options of the suppression tank (ST) with or without water as heat sink, and impact of the cooling ability of the vacuum vessel (VV) have been investigated. MELCOR 1.8.6 for fusion is a valid tool for this study. The ex-vessel LOCA is initialized during the normal operation at the steady state. The transient results of different scenarios will be discussed for the time evolution of the accident sequences, pressurization in the TCR, VV and ST, temperature behavior in different volumes and structures, and tritium and dust transport behavior.

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High temperature brazing of tungsten with steel by Cu-based ribbon filler alloys

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The work presents the results of high temperature brazing of tungsten with EK-181 steel by rapidly quenched into ribbon filler alloys based on copper. Compositions of the filler alloys were chosen with consideration to the requirement of reduced activation that is necessary for DEMO reactor. All the joints were manufactured at 1100oC in a vacuum furnace. To analyse microstructure and mechanical characteristics before and after thermocycling tests (in the interval of 700 to 25 oC) SEM investigation, microhardness and shear strength tests were used. It is stated, that the use of Cu-27Ti and Cu-20Sn filler alloys together with vanadium interlayer gives an opportunity to make qualitative joint.

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Interface and requirements analysis on the DEMO Heating and Current Drive system using systems engineering methodologies

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In this paper we present the analysis of System Requirements and Interfaces of the Heating and Current Drive (HCD) system of the Demonstration Fusion Power Reactor DEMO.
The work was performed applying Model-Based Systems Engineering (MBSE) refining the HCD System Architecture for assessing the system functions, its interdependencies and its overall integration into DEMO. Two concepts for DEMO have been considered: a conservative design (DEMO1) and a more advanced one (Flexi-DEMO). The effort has been undertaken in the frame of the Work Package Heating and Current Drive, supported by the Work Package Plant Level System Engineering, Design Integration and Physics Integration.
The scope of the work is, on the one hand, to address the identification and definition of the interfaces occurring, both internally in the HCD system, and between the HCD system and neighboring systems. On the other hand, the impact of requirements coming from the ongoing physics studies has been assessed.
The rationale is to provide the technical foreground for supporting the decision-making processes related to the HCD system which is planned to be carried out during the Conceptual Design Phase, currently foreseen until 2027. The results we show in this paper are part of the design and integration activities consisting of both systems engineering methodologies and design analysis, all aiming at ensuring consistency in the overall EU DEMO plant design. The set of processes we report here is in line with the common approach established within PPPT and in accordance with the ISO Systems Engineering standard (ISO 15288).

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

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Platinum suported on graphene - PTFE as catalysts for isotopic exchange in a detritiation plant. Manufacturing and physical and

microstructural analysis

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The catalytic separation of hydrogen isotopes is of particular interest for nuclear industry from the point of view of tritium recovery and its use in fusion reactors. Isotopic exchange may take place in the homogeneous (gaseous) phase or in the heterogeneous phase (hydrogen or gaseous deuterium and water or liquid heavy water). Catalysts are necessary both for the homogeneous phase reaction and the heterogeneous reaction.
Recently, graphene and graphene oxides have been investigated as support material due to their excellent physical and chemical properties. The high specific surface, their thermal and chemical stability, high mechanical strength make graphene a potential component in the development of new catalysts that could be used in isotopic exchange. The use of graphene and graphene oxides as support for the active metal has shown an improvement in catalytic activity when compared to the conventional support, degasolination carbon. This behaviour may be explained by the high dispersion of active metal on the surface of graphene.
The paper presents the preparation of some catalysts that will be used in the future catalytic isotopic exchange experiments (in the beginning VPCS in order to understand its efficiency in isotopic exchange then the catalysts will be integrated into a LPCE-type laboratory installation). The catalysts thus prepared will be characterised from the microstructural point of view for a comparison with existing catalysts. BET, SEM, TGA and XRD will be used for the physical and microstructural analysis of catalysts.

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Alloy element induced vacancy clustering in W-Re/Ta material

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Alloying elements can possibly serve as an important technique in designing W-based plasma facing materials (PFMs) with superior comprehensive performance. To investigate the interaction between the alloying elements and point defect is one of the mainly contents to study the W material service properties under irradiation. The first-principle method based on the density function theory was used to investigate the laws for the movements of the point defects in case of the alloying metal Re/Ta solution atoms. It was found from the formation energy and binding energy that Rhenium would be a potent factor to form the mono-vacancy than Ta in W alloying system. The mono-vacancy diffusion in W-Re/Ta alloy was studied via the pathways models, in which Re atom can steady exist in tungsten lattice while Ta solute atom was beneficial to move to a vacancy. Although the replacement atoms were different, it could find that both solute Re and Ta atoms manifest the positive effects to mono-vacancy migration in irradiation environment. According the average relaxed volume, the attraction interaction of vacancy cluster can be interpreted rationally. The important thing was that Re contributed the vacancy to form clusters configuration, but Ta had an inhibiting effect on diminutive vacancy cluster ($n \leq 3$), and it can be furthered concluded that Ta could control the point defects from gathering together, especially when its concentration was enlarged. These results can be used to construct the design basis of the W-based alloying in terms of improving the radiation resistance in the fusion environment.

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mitigation of an ingress coolant event in ITER vacuum vessel by means of steam pressure suppression

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An Ingress of Coolant Event (ICE) is postulated to occur in the ITER Vacuum Vessel (VV) due to a breach on the first-wall cooling channels. The pressure raise in the VV is limited by means of a Vacuum Vessel Pressure Suppression System (VVPSS), consisting of relief lines connected to the VV and discharging the steam to four Vapor Suppression Tanks (VST) partially filled with water: one managing small water leaks and three together managing large water leaks).
The ITER vapor suppression system operates at sub-atmospheric pressure. Since no detailed information has been found in the technical literature for similar operating conditions, a specific study has been launched at the University of Pisa to investigate the performance of the vapor suppression process in the VSTs.
Both an analytical-numerical analysis and an experimental activity were carried out on a 1:21 scale apparatus, simulating the VVPSS.
Scaling studies were first performed in order to calculate the transient of steam mass flow rate occurring during the ICE IV event in the scale apparatus, starting from the thermal hydraulic study performed at ITER. Subsequently, a quite extensive experimental campaign was carried out on the scale apparatus in order to study the influence of the main thermal hydraulic parameters that characterize the steam condensation efficiency in sub-atmospheric conditions.
This paper illustrates the results obtained from the experimental transient simulating the Ingress of Coolant Event in the VST Tanks.
The experimental results confirm the determined scale predictions and the capability of the VVPSS tank to condense the injected steam at sub-atmospheric pressure, matching the safety goal to limit the VV pressurization.

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THE HIGH VOLTAGE DECK 1 AND BUSHING FOR THE ITER NEUTRAL BEAM INJECTOR: INTEGRATED DESIGN AND INSTALLATION IN MITICA EXPERIMENT

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MITICA is the full scale prototype of ITER Heating Neutral Beam (HNB), designed to deliver 16.5MW of heating power to ITER plasma, currently under construction at the Neutral Beam Test Facility in Padova (Italy). In ITER HNB, negative ions (H-/D-) are produced in the Ion Source (IS) polarized to ground at -1012kV, then extracted by 12kV extraction voltage, accelerated to ground at 1MeV energy and finally neutralized. The complex power supply system feeding the IS includes two non-standard equipment, beyond the actual industrial standard for insulation voltage level (-1MVdc) and dimensions: 1) the HVD1 (High Voltage Deck1), a large air insulated Faraday cage (12.5m (L) x 8.4m (W) x 9.6m (H)) hosting the IS power supplies and diagnostics; 2) the HVBA (High Voltage Bushing Assembly), a -1MVdc air to SF6 Bushing which interfaces the HVD1 with the SF6 insulated Transmission Line (TL) connecting the Acceleration Grid Power Supply system (AGPS) with the IS, carrying inside its High Voltage electrode all IS power and diagnostics conductors. The HVD1 and HVBA are installed inside a large HV hall, with controlled ambient conditions. The main design choices leading to the final hall layout, integrating the HV experimental equipment and the conventional building plants by assuring the necessary clearance required by such very high electric insulation level, are presented. Moreover, the paper presents the manufacturing design developed by the supplier to meet the design constraints and requirements of the technical specification for such unconventional devices. The factory type tests to validate the design and release the manufacturing of the HVD1 are described together with the electrical, mechanical and thermal tests carried out on the HVBA.

Finally, the paper reports on the on-site installation and commissioning and testing activities carried out in 2017 and on the final acceptance at full voltage foreseen in 2018.

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Technical proposals for the IGNITOR control, data access and communication system

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The control, data access and communication system (CODAC) designed to solve the tasks of planning, preparing and conducting the experiment, collecting, processing and complex analysis of the experimental results at the IGNITOR tokamak fusion project.
It is proposed to build CODAC based on modern failsafe dual-redundant industrial equipment manufactured by National Instruments, Schneider Electric, Siemens, Highland Technology, Hewlett-Packard, GE Fanuc etc. Cross-system interaction provided by high-speed fiber-optic networks: Reflective Memory, MXI-Express, Ethernet network and pulse synchronization with a few microseconds time delay. The high level of Fast Systems (the control loops above the 100 Hz) based on Mathlab Simulink RT (Linux operation system) and National Instruments Labview RT: plasma control system, central safety system, coil power supply system, diagnostics etc. The high level of Slow Systems (the control loops are slower or equal of 100 Hz) based on Wonderware InTouch: auxiliary process systems i.e. vacuum system, cooling water system etc. IGNITOR CODAC should provide synchronized real-time high throughput data processing, high reliability, availability, maintainability and access to experimental data for remote participants through the unified protocol via Internet.

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Structural integrity assessment of an ITER ECH&CD Upper Launcher mirror (LM1)

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The Lower Mirror one (LM1) is part of the in-vessel quasi-optical beam propagation system for the ITER Electron Cyclotron (EC) Upper Launcher (UL), in which each of eight beams of mm-waves are reflected from four mirrors during passage to the plasma. 60000 thermal cycles are foreseen at frequencies lower than 3Hz and power levels up to 1.31MW per beam.
This paper reports the means used to accurately ascertain resistance against the following failure modes: 1) plastic collapse, 2) thermal fatigue and 3) ratcheting. Analyses are carried out in accordance with the design-by-analysis approach. Transient thermo-mechanical effects are investigated via finite elements to support the assessment during the thermal cycle.
This structural integrity study of the LM1 mirror using the American code and standards for pressure vessels and piping aims to confirm the compliance of the proposed design.
This work was supported in part by the Swiss National Science Foundation.
This work was carried out within the framework of the ECHUL consortium, partially supported by the F4E grant F4E-GRT-615. The views and opinions expressed herein do not necessarily reflect those of the European Commission or the ITER Organization.
Keywords: LM1, Upper Launcher, Transient Thermo-Mechanical Analysis, Structural Integrity, Fatigue

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Implementing DevOps practices at the control and data acquisition system of an experimental fusion device

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The stellarator Wendelstein 7-X (W7-X) is a fusion device designed for steady state operation. It is a complex technical system. To cope with the complexity a modular, component-based control and data acquisition system has been developed.
During operation phases of W7-X components steadily evolve. For instance measurement devices for
diagnostics get improved, technical processes are optimized, experienced limits of the machine have to be taken into account or simple "bug fixing" is done. This requires continuous further development of the components while operating them at W7-X – a typical use case for DevOps practices.
DevOps is a software engineering practice. The term is a compound of development and operations. It aims at shorter development cycles, while the quality of the changed system must stay at a high level. This is achieved by using a highly automated tool chain.
DevOps at W7-X does not (only) mean deploying new software packages. It also comprises setting up
new configurations, changing rules and constraints, improving experiment planning views etc. of components, based on new scientific and technical experiences. One crucial step of the DevOps tool chain at W7-X is the release and reconcile process. It is a well defined and automated process that commits a change in the components configuration (Release) to the control and data acquisition system with minimal impact. During the process all existing experiment programs are adapted to this change (Reconcile). For newly added configuration parameters default values are added. Thus it
is guaranteed that experiment programs are still executable at W7-X. The quality is ensured by preceding unit and integration test, configuration consistency and version checks.

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T-15MD tokamak plasma control platform architecture

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A key feature of the developed T-15MD tokamak Plasma Control System (PCS) is its ability to rapidly design, test and deploy real-time shot scenario algorithms. PCS platform consist of two levels:
1. High application-specific level: model development and linear approximation, calculation of the experiment scenario, controllers design and experiment simulation (Mathlab Simulink RT high-performance computer);
2. Process control level: real-time control of plasma parameters (National Instruments hardware running LabView RT operating system).
In the Hardware-in-the-Loop simulation mode communication between the levels (1) and (2) is realized by the reflective memory (RFM) "star" topology network and the middleware S-function package within Simulink RT environment. RFM providing a deterministic real-time method of sharing memory between the systems.
The proposed architecture of the platform allows to perform testing and configuration of the PCS before plasma shots, which increases the efficiency of the experiments while reducing costs. In future, the plan is to use the high-performance computer Simulink RT to perform real-time calculations

of plasma reconstruction and equilibrium in the magnetic control loop, implementing PCS data exchange in the RFM network. It is planned to develop infrastructure for simplified integration and testing of third-party control algorithms and plasma-physical codes.

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Development of medium size DOME & reflector plate for ITER like tokamak application

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A medium sized water-cooled Divertor DOME has been manufactured at Institute for Plasma Research (IPR), India. Divertor Plasma Facing Components (PFCs) such as DOME and Reflector Plate has multi-layered joints which are made of various materials such as Tungsten (W), OFHC Copper (Cu), Copper alloy (CuCrZr) and SS316L etc. Joining of such multi-layered joints is known to be problematic as being used of several dissimilar materials. Vacuum brazing route was employed to fabricate the medium size DOME as well as the Reflector plate. In order to evaluate the performance of the DOME against ITER-like scenarios, the DOME has been successfully tested for 1000 numbers of steady-state thermal cycles with an incident heat flux of 3.87 MW/m2 in the High Heat Flux Test Facility (HHFTF) at IPR. Subsequent testing with additional 200 thermal cycles was also done with an incident heat flux of approx. 6 MW/m2 at 24.6 kW of Electron beam power. The absorbed heat flux was calculated to be 4 MW/m2 which indicated the absorbed power was nearly 70%. During the HHF tests, the surface temperature of the W tile reached 640oC and the beam power was restricted to 24 kW due to the temperature limit of 450oC at the CuCrZr heat sink. A total 1200 cycles of steady-state thermal cycles have been completed. Engineering analysis on the HHFT of the DOME has been performed using Finite element method (FEM) and Computational Fluid Dynamics (CFD) to simulate and to correlate with the experimental data.
Ultrasonic immersion technique (NDT) was incorporated to inspect the brazed joint quality of the Reflector plate and the DOME mock-up before and after the HHFT. The results of the experimental details, engineering analysis and methodology adopted to fabricate the DOME and reflector plate will be presented in the paper.

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Progress in Development of ITER Diagnostic Pressure Gauges and Status of Interfaces with ITER Components

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Neutral gas pressure is one of the main parameters for basic control of ITER operation. Diagnostic Pressure Gauges shall provide pressure measurements in the range from 10-4 Pa to 20 Pa with an accuracy of 20 % and a time resolution of 50 ms. In total 52 DPG sensor heads will be installed in 4 lower ports, 4 divertor cassettes and 2 equatorial ports. The overall DPG system has 15 interfaces with various systems and components of ITER.
Within a Framework Partnership Agreement with F4E, IPP is developing the DPG system including

sensor head and front-end electronics. A test campaign aiming at validation of the system baseline design is currently ongoing.
Performance of the DPG sensor head has been investigated for various parameters of the prototype as electrode potentials, transparency of the acceleration grid and electron emission current. The test results demonstrate the possibility to measure pressures up to 30 Pa.
Emission properties of several material candidates for the filament (hot cathode) have been studied during approximately three months of continuous operation. Filament samples made of Tungsten with 2 % doping of ThO2 and Tungsten alloy with 26 % of Rhenium coated with Y2O3 demonstrated most promising results. These cathodes required lowest heating currents for fixed emission current. Mechanical tests of these samples showed no considerable deformations for maximum heating currents in transient magnetic fields of up to 8 T.
The already obtained as well as future test results will be used for further design optimization and integration of the DPG system into the ITER environment.

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Design of scalable vacuum pump to validate sintered getter technology for future NBI application

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The vacuum systems of neutral beam injectors have very demanding requirements in terms of gas type, pumping speed and throughput. Due to its high affinity to hydrogenic species, non-evaporable getter (NEG) is in principle a good pumping technology candidate for the deployment in neutral beams, which require the injection of a serious amount of hydrogen in order to operate. Getter materials operate at room temperature, and their use could be particularly welcome in the absence of cryogenic supplies, if high temperature superconducting magnets are successfully deployed in future fusion plants. In the past NEGs have not be used due to their insufficient capacity, but with the new materials developed in recent years, a big step forward has been done.The strategy to validate the use of getter pump technology is based on the realization of a relatively large pump mock-up that has to be tested in fusion-relevant conditions. The objectives of this mock-up are to demonstrate that a pump of large dimensions and capacity is usable. This paper deals with the design of the mock-up, based on conceptual studies which involve at first 3D gas flow simulations considering different modular mock-up pumps based on NEG sintered disks. In addition transient thermal simulations with FE method have been performed with the aim to analyze the thermal response of the mock-up. The conceptual design has been carried out in order to define the best configuration to obtain high pumping speed with low spatial gradient of gas concentration inside the getter material. The suggested solution will exhibit a modular structure of getter disks, which on one hand simplifies the mechanical assembling, and on the other hand allows interpretative modelling at different scales. It is foreseen to test the pump mockup in the TIMO facility at KIT Karlsruhe.

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Work-flow process from simulation to operation for the Plasma Control System for the ITER First Plasma.

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The ITER Plasma Control System (PCS) is an essential component for ITER operations. It will include multiple controls loops as well as a number of support functions dedicated to providing input control parameters and distributing commands to actuators. In addition, a supervisory system within the PCS architecture will manage the orchestration of the PCS control loops during the discharge as well as an Event Handling system to react to real-time changes in the plasma and/or in the plant systems. To investigate and develop controllers and supervisory techniques, a simulation platform is being developed aiming at supporting the staged evolution of the PCS. The PCS Simulation Platform (PCSSP), based on Matlab/Simulink, is used to design and assess PCS control functions and verify the consistency of the PCS architecture. Moreover, the PCSSP is planned to assist in the implementation of PCS algorithms, directly deploying PCS code on the ITER Real Time Framework (RTF).
This paper will present the overall work-flow processes from control simulations, control assessment, code generation, interfaces and deployment into the RTF and PCS commissioning. This will include interface management and co-ordination with ITER investment protection, the pulse scheduling system, and the plant system configuration during operations. How the entire preparation process will be managed, for ITER first plasma operations, will be reported here.

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Development of plasma control algorithm design via machine learning

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Machine learning has garnered increasing attention within the fusion community in recent years, with much of the focus going toward implementation of disruption predictors. However, disruption detection is but one possible area in which the large body of fusion experimental data, accrued over decades, can be put to use. In particular, this data can be utilized to assist in the implementation of closed loop controllers, either through augmentation of existing model-based approaches, or via purely data-driven methodologies.

In this work we explore the use of machine learning for vertical stability control of the DIII-D tokamak. We describe the application of a search and machine learning computing toolchain for system identification of highly non-linear coil/vessel/plasma interactions as a function of equilibrium state. Extraction of the training data used in this process is achieved at rates over two orders of magnitude greater than previously attainable. The identified system model is then integrated with a model predictive controller and tested in simulation. Additionally, we investigate creation of a purely data-driven vertical control algorithm (using, for example, reinforcement learning). Development toward integration of these algorithms for real time use in the DIII-D plasma control system is discussed. The use of large-scale machine learning techniques for plasma control is still novel within the fusion community, and this work provides a template for future data-driven approaches.

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Development of data acquisition and control system for quasi-2D turbulent electrolyte flow experiment

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A new experimental device has been designed, manufactured and tested for quasi-2 dimensional turbulence studies in magnetized electrolyte system. This experiment can provide significant information about the interaction of large scale shear flows (zonal flows) and smaller scale turbulent vortices. This physics problem has a relevance in different scientific areas such as the turbulent transport reduction via shear flows in magnetically confined fusion plasmas.

14x14 pieces of N52 10x10x10 mm neodymium magnets have been placed below a plastic container in different geometrical configurations in order to create a large variety of turbulent drives together with a few amps DC electrical current which has been driven through the electrolyte (NaCl). Beside the small vortices induced by the permanent magnets, a large (system-size) seed flow has been also generated by a solenoid placed below the flow. Using this set-up, the interaction of large sheared flows with small scale vortices can be experimentally studied.

This paper describes the whole system: the hardware and the software developed for the turbulent flow experiment described above. In order to evaluate the velocity field of the flow observed by a high resolution camera, PIV (Particle Image Velocimetry) techniques have been used. Matlab, Simulink, sensors and actuators with microcontroller for data acquisition and process control have been implemented. We also developed our own Matlab routines for automated evaluation of the measurements, enabling remote (overnight) data processing.

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Design and preliminar operation of a laser absorption diagnostic for the SPIDER RF source.

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The ITER Heating Neutral Beam (HNB) injector is required to deliver 16.7 MW power into the plasma from a neutralised beam of H-/D- negative ions, produced by an ICP RF source and accelerated up to 1 MeV. To enhance the H-/D- production, the surface of the acceleration system grid facing the source (the plasma grid) will be coated with Cs because of its low work function. Cs will be routinely evaporated in the source by means of specific ovens. Monitoring the evaporation rate and the distribution of Cs inside the source is fundamental to get the desired performances on the ITER HNB.
In order to proper design the source of the ITER HNB and to identify the best operation practices for it, the prototype RF negative ion source SPIDER has been developed and built in the Neutral Beam Test Facility at Consorzio RFX. In SPIDER, the dynamics of Cs will be monitored by measuring the emission intensity of the Cs 852 nm line along several lines of sight in the source. For a more quantitative estimation of Cs density a Laser Absorption Spectroscopy diagnostic will be installed; by using a wavelength tunable laser, the diagnostic will measure the absorption spectrum of the 852 nm line along 4 lines of sight, parallel to the plasma grid surface and close to it. From the absorption spectra the line-integrated density of Cs at ground state will be calculated.
The paper will present the design of the diagnostic for SPIDER, with a description of the layout and of key components. The paper will also show the first experimental results from a preliminary installation of the diagnostic on the test stand for Cs ovens.

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Design and development of the mechanical support structure for ITER in-vessel magnetic sensors

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A mechanical support structure (a.k.a. "platform") has been designed to provide mechanical support and thermal conductance for the inductive magnetic sensors installed on the inner shell of ITER vacuum vessel (VV) for equilibrium and high-frequency magnetic field measurement. The platform design is modular so as to simplify the on-site installation process. It consists of a permanent and a removable section. The permanent section is welded to the VV surface, forming the base of the whole assembly and allowing electrical connection with the in-vessel electrical service. The removable section houses the sensor head, which is engaged with the permanent base through a knife-switch type connector. Alignment of the two sections is realized via two alignment pins built into the permanent base. The removable section is designed to be compatible with remote handling operation in case of replacement. Real-scale prototypes of the platform components had been successfully manufactured and assembled together according to the design. Good alignment between the two sections has been achieved. Full electrical continuity from the sensor head down to the electrical service cabling has been demonstrated in the platform prototypes. Continued R&D efforts will focus on testing some installation and operation aspects, including mechanical vibration, thermal cycling, cable clamping, etc.

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Conceptual studies on optical diagnostic systems for plasma control on DEMO

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The roadmap to the realization of fusion energy describes a path towards the development of a DEMO tokamak reactor, which is supposed to provide electricity into the grid by the mid of the century [1]. The DEMO diagnostic and control (D&C) system must provide measurements with high reliability and accuracy, constrained by space restrictions in the blanket under the adverse effects induced by neutron, gamma radiation and particle fluxes. As a consequence an initial selection of suitable diagnostics has been obtained [2]. This initial group of diagnostic consists in 15 different systems classified in 6 methods, microwave diagnostics, thermo-current measurements, magnetic diagnostics, neutron/gamma diagnostics, IR interferometry/polarimetry, and a variety of spectroscopic and radiation measurement systems.

The key aspect for the implementation, performance and lifetime assessment of these systems on DEMO is mainly attributable to their suitable location, that must be well protected against neutrons, and meet their own set of specific requirements. Within this paper, we concentrate on spectroscopic and radiation measurement systems that require sightlines over a large range of plasma regions and inner reactor surfaces. In this context, sightline analysis, the space consumption and the evaluation

of optical systems are the main assessment tools to obtain a high level of integration, reliability and robustness of all this instrumentation, essential features in future commercial fusion power nuclear plants. This paper summaries the main results and strategies adopted in this early stage of DEMO conceptual design, to assess the feasibility of this initial set of diagnostic methods based on sightlines and the integration of the total number of these needed for DEMO D&C.

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EHF FW panel for ITER BM with mechanical attachment of the plasma-facing components

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The qualification Full Scale Prototype (FSP) of the Enhanced Heat Flux (EHF) First Wall (FW) for the ITER blanket will be manufactured by JSC NIKIET and JSC NIIEFA as part of the Procurement Arrangement between the ITER Organization and the Russian Federation Domestic Agency. The FSP design is based on the FW panel #14 type A and includes: the supporting structure (FW beam), plasma facing components (FW fingers) and the central slot insert (CSI) to protect the FW beam/fingers welded joint from plasma-induced heat flux.
In 2017 the EHF FW panel design has been revised in order to improve the repairability and to provide non-destructive inspection of the FW components during the final assembly stage. The modifications include replacing the FW beam/fingers welded joint by a mechanical attachment. The mechanical attachment is located under the FW fingers and is therefore shielded from plasma-induced heat flux. This also allows the possibility of excluding the CSI from the FW panel design which would simplify the cooling channel layout; however the thermal load on the middle part of the FW beam in the absence of the CSI needs to be further assessed before a decision can be taken.
In this connection JSC NIKIET specialists have performed parametric analyses of this FW panel design, to investigate the effect of the depth of the slot in the middle of the panel.
This paper describes the revised design of the EHF FW Panel with a mechanical attachment of the plasma facing components to the FW beam, and discusses the results of the corresponding thermal and structural analyses.

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Inverse heat flux evaluation of STRIKE data by neural networks

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The instrumented calorimeter STRIKE (Short-Time Retractable Instrumented Kalorimeter Experiment) has been designed with the main purpose of characterizing the SPIDER negative ion beam in terms of beam uniformity and divergence during short pulse operations. STRIKE is made of 16 1D Carbon Fibre Composite (CFC) tiles, intercepting the whole beam and observed on the rear side by infrared (IR) cameras.
The front observation presents some drawbacks due to optically emitting layer caused by the excited gas between the beam source and the calorimeter, and the material sublimated from the calorimeter surfaces due to the heating itself. It is then necessary to solve an inverse non-linear problem to determine the energy flux profile impinging on the calorimeter, from the 2D temperature pattern measured on the rear side of the tiles. Most of the conventional methods used to solve this inverse problem are unbearably time consuming, so a ready-to-go instrument to determine the beam condition, while operating STRIKE, is mandatory. In this work, the inverse problem, both in stationary and non-stationary conditions, is faced by using a Neural Network (NN) model, pursuing two different approaches. In the first one, the NN is trained to directly solve the inverse problem, by associating the radiation profile (target) to the measured temperature profile (input). In the second approach, a NN is trained to solve the direct problem, where the input is the radiation profile and the target is the temperature profile. Then, the NN is inverted by determining the input corresponding to a fixed target. Preliminary results show the reliability of the proposed method for STRIKE real time operation.

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Plasma light detection in the SPIDER beam source

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The ITER Heating Neutral Beam (HNB) injector RF plasma source is required to generate a 40A D- or 46A H- ion current, with low electron/ion ratio (<1) and high uniformity over the extraction area (800 mm x 1600 mm). The source prototype SPIDER in the Neutral Beam Test Facility at Consorzio RFX has been developed to demonstrate these performances and it is now under final installation and commissioning.
A set of diagnostics in SPIDER has to characterize the complex behaviour of the plasma in the source, assisting in the optimization of performances and providing the signals for machine protection during operation. To monitor the visible radiation emitted by the plasma in the source a set of photodiodes measures the time evolution of lines of sight integrated intensity; some are equipped with interference filters to select specific spectral lines of particular interest, like of cesium to control its evaporation, which assists the H- production, and of metal impurities lines, which are sign of severe erosion of the source inner wall or even of melting at hot spots. Other lines of sight are fed to grating spectrometers to measure the spectrum at a slower rate, either for low resolution spectral survey of hydrogen Balmer lines and impurities and for high resolution molecular Fulcher bands.
This contribution focuses on the implementation of the photodiode measurements and on preliminary results during the commissioning phase. The overall diagnostic layout is presented. The custom photodiode electronics is described, which includes a low noise variable gain amplifier, FPGA based remote control and suitably shaped digital output for interlock purposes.

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Optimization of GEM based detector structure aimed at plasma soft–semi hard X-ray radiation imaging

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The proposed work refers to the development of gaseous detectors for application at tokamak plasma radiation monitoring. Soft–semi hard X-ray region radiation measurement of magnetic fusion plasmas is a standard way of accessing valuable information on particle transport and magnetic configuration.
In this work, Gas Electron Multiplier (GEM) based imaging technique is proposed to perform advanced imaging, capable of photon energy discrimination, which can reach a very accurate spatial and temporal resolution and can provide lots of information on radiation, temperature and impurity distribution, MHD, etc., including data processing on the fly (in real-time) usable for plasma control purposes.
The work will highlight the latest development and optimization on GEM detector structure for utilization in soft–semi hard X-ray radiation detecting system. The photon sensitive configuration is based on triple GEM amplification structure followed by the pixel readout electrode. The efficiency of detecting unit is adjusted for the radiation region of about 2-15 keV. The present work will introduce the preliminary laboratory results on the detector characteristics obtained for the constructed detecting chamber based on newly designed and developed GEM foils. The operational characteristics and capabilities of the detector will be compared with the ones based on standard (commonly used) copper GEM foils. First laboratory tests will be done by means of tuned X-ray laboratory source (2-9 keV), commercial X-ray tube (10-13 keV) and 55Fe iron sources presenting detector's imaging capabilities. Stream-handling data acquisition mode will be applied for the data acquisition with timing down to the ADC sampling frequency rate (~13 ns) which allows uncovering the invaluable physics information about plasma dynamics due to excellent time resolution. The spatial resolution and imaging properties of this detector will be studied in this work for conditions of laboratory moderate counting rates and high gain.

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Protection of window assemblies against ECRH and CTS stray radiation in ITER

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Stray radiation at 60GHz and 170GHz is an engineering challenge for the integrity of various window assemblies in ITER. Their protection and long term performance preservation are essential for both the operational safety of the device and its scientific exploitation. This contribution focuses on the assessment of Electron Cyclotron Resonance Heating (ECRH) and Collective Thomson Scattering (CTS) stray loads impacting the windows of one of the most affected diagnostic port plugs: #11. A Neutral Particle Analyzer (NPA), the Low Field Side Reflectometer , Vacuum Ultra Violet (VUV) and Halpha spectrometers are connected to this port plug and may need protection. Indeed, as ECRH/CTS stray radiation may last throughout most of the discharge, an increase in window

temperature may occur leading to excessive stresses in the assemblies. Appropriate measures have therefore to be taken to improve the attenuation of their transmission lines. The attenuation of both at 60GHz (CTS) and 170GHz (ECRH) frequencies has been evaluated with both CST simulations and analytical calculations. The use of mitigation measures (such as coatings, grids, etc) has been considered. Solutions with quite good perspectives have already been identified for the NPA, VUV and Halfa lines. Further studies are under way for the design of the Low Field Side Reflectometer, considering active and/or passive components to dampen the stray radiation travelling through the line. Finally, a preliminary set of qualification tests both for the coating materials and for the fused silica window assemblies, with particular attention to the measurement of their dielectric properties, will be discussed.

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Exploring the upper measuring limit of pressure gauges for ITER by experimental variation of instrumental parameters

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Neutral gas pressures in the vacuum vessel of ITER will be measured by hot cathode ionization gauges. The design is based on the ASDEX pressure gauge which is operated successfully in many fusion experiments worldwide. Further development is needed to fulfill superior requirements: the upper measuring limit has to be at least 20 Pa in hydrogen at a magnetic flux density of up to 8 T. The required measurement accuracy of 20 % implies sufficient differential sensitivity.
This work aims at proving compliance of the pressure gauges with requirements of the ITER experiment by means of laboratory tests.
An experimental campaign was conducted to explore the accessible measuring range by consecutive variation of gauge parameters: electron emission current, electrode potentials and transparency of the acceleration grid. A special prototype was manufactured that features an exchangeable acceleration grid; transparencies from 20 % to 80 % are available. The ion over the electron current as a function of pressure and magnetic flux density was obtained for each parameter set.
A monotonic behavior was achieved even up to 30 Pa by a reduction of the grid transparency and an increase of the electric field strength at the cathode to accelerate the electrons. While the former leads to a lowering of the ion current, which is unfavorable for the sensitivity limit in the low pressure range, this can be compensated in part by a higher electron current.
The reproducibility of the gauge response under repeated experimental conditions is to be tested in near future. These results enable the calculation of statistical errors to estimate the measurement accuracy.

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Progress in the production of the W7-X divertor target modules

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The realization of the 19.6 m² highly heat loaded surface of the actively water-cooled divertor of Wendelstein 7-X (W7-X) requires the installation of 100 target modules distributed in ten discrete similar divertor units. A target module is made of target elements mounted onto rails joined by a stiffening plate forming a frame with an attachment system to the plasma vessel. The target modules are water-cooled from manifolds to distribute the water equally between the target elements with flanged connections to the water supply system. A target element is made of a CuCrZr copper alloy heat sink armored with carbon fibre reinforced carbon CFC NB31 tiles and designed to remove a stationary heat flux up to 10 MW/m² on its main area. The main challenge was to fit modules in the limited available space taking into account the 3-D shape of the plasma vessel, the neighboring in-vessel components and diagnostics. The assembly of the modules was carried out in the workshop of IPP-Garching. Special attention was given to the positioning of the individual target elements onto the supporting frame to avoid local heat loads or leading edges, and on the reliability of the orbital welding for the cooling circuits. The quality of the target modules was assessed as follows: visual inspections, measurement of the 3-D CFC surface, dynamic pressure tests, He leak testing under pressure at different temperature (20℃, 160℃) in vacuum oven, high heat flux testing. The paper presents the design, manufacturing process and the results of the quality assessment of the 30 first finished modules to be positioned in the horizontal part of the divertor.

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Optimization of single crystal mirrors for ITER diagnostics

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Diagnostic mirrors are planned to be used as plasma-viewing optical elements in all optical and laser-based diagnostics in ITER. Degradation of mirrors due to e.g. deposition of plasma impurities will hamper the entire performance of affected diagnostics. In situ mirror cleaning by plasma sputtering is presently envisaged for the recovery of optical reflectivity of contaminated mirrors.
Previous studies have demonstrated sound advantages of single crystal mirrors, outlining their ability to withstand plasma sputtering without noticeable degradation of optical reflectivity. At the same time, there are studies made on polycrystalline substrates, showing a signature of sputtering dependence on crystal orientation. Should such a dependence exist, the sputtering of single crystals could be minimized, thus prolonging a mirror lifetime in ITER.
Four single crystal (SC) molybdenum (Mo) mirrors with different crystal orientation were produced to study the effect of crystal orientation on sputtering. Mirrors were exposed to steady-state argon plasma in linear plasma device PSI 2 under identical plasma conditions. The energy of impinging ions was about 60 eV. Mirror temperature was 250oC. Sputtering conditions in PSI 2 were corresponding to those expected inside the mirror cleaning system of ITER. The average amount of sputtered mirror material was about 1200 nm corresponding to about 120 mirror cleaning cycles.
Plasma exposures did not affect the optical performance of all mirrors. The maximum measured decrease of specular reflectivity did not exceed 5% in the ultraviolet range at the wavelength of 250 nm. No increase of the diffuse reflectivity was detected. The single crystal mirrors with orientations [110]/[101] demonstrate at least 25% less removal of sputtered material than mirrors with other crystal orientations. Summary of results, their analysis will be presented along with a feasibility assessment of the use of optimized mirrors in ITER.

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Progress in high heat flux testing of European DEMO

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In the framework of the DEMO divertor project of EUROfusion an extensive R&D program has been carried out to develop advanced design concepts for hot water cooled divertor targets. These plasma-facing components made of W blocks as plasma facing material and CuCrZr tubes as cooling tubes should allow a reliable DEMO operation for 2 h long pulses and maximum heat fluxes up to 20 MW/m². Compared to ITER, the operation at higher coolant temperature of 150 ℃, the longer required lifetime, and the significantly higher neutron fluence are the design challenges exceeding the current extent of experience.

In the pre-conceptual design phase, eight types of target mock-ups were designed and manufactured by the involved research groups. Finally, 25 of these unirradiated mock-ups were assessed by high heat flux (HHF) examination in the test facility GLADIS at IPP Garching. The applied hot water cooling at 130℃ inlet and 16 m/s velocity ensures thermal conditions similar to the expected DEMO operation. To reduce the experimental effort of the HHF testing each individual mock-up was subjected to a screening test up to 20 MW/m² at 20℃ cooling water inlet for first selection, followed by HHF fatigue tests between 10 and 20 MW/m². After the cold water HHF testing, six concepts were qualified for the subsequent hot water HHF testing with 300 cycles at 20 MW/m². Four of them were successfully loaded up to 500 cycles. Extensive diagnostics such as high-resolution infra-red and visible light cameras were employed for in-situ assessment of surface temperature evolution during the cyclic HHF loading.

This contribution presents the summary of HHF testing and results from the post exposure investigation. Overall HHF performance of the various design concepts will be discussed together with other non-destructive investigations including ultra-sonic examination and transient IR thermography.

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Endoscopes for observation of plasma-wall interactions in the divertor of Wendelstein 7-X

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The stellarator Wendelstein 7-X has been prepared for long pulse operation in the first operational campaign. Forschungszentrum Jülich has contributed diagnostics for investigation of plasma wall interaction processes in presence of an island divertor and steady state plasma at high density and low temperature.
A versatile optical observation system has been developed for local characterization of the divertor plasma and the divertor target surface. At two opposite divertor locations linked by magnetic field lines, temperature and density profiles, impurity transport and recycling, hydrogen recycling, and transient heat load deposition are analysed. The analysis is assisted by two gas inlets in the divertor,

a fast probe manipulator and a poloidal correlation reflectometer located along the same magnetic structure.
The optical systems consist of two endoscopes each with perpendicular fields of view with the opportunity of tomographic reconstruction. Mirror based optics has been chosen in order to be independent of wavelength. A narrow field of view allows for high spatial resolution while rotation of the first mirror covers the full poloidal divertor sections. An integrated shutter mechanism and a vacuum window far back minimize coating of optic components. For assessment of change of transmission, a relative calibration function is implemented. The output light is split into wavelength ranges. Both, cameras equipped with narrow band filters as well as spectrometers are connected. Six cameras and four spectrometers can be attached simultaneously to each endoscope.
In this contribution, the detailed optics, mechanic, and thermal concept, experience from assembly and adjustment as well as first results will be presented.

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Thermal Induced Electromotive Force Measurements On Twisted Pair Mineral Insulated Cables

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The components of the ITER Diagnostics are located all over on the inner and outer shell of the Vacuum Vessel, in the Ports, on the Divertor Cassettes and in the Cryostat as well. Sensors require electrical transmission lines to transmit both of the diagnostic and control signals across the vacuum boundaries. To transmit the signals, Mineral Insulated cables will be used.
During the last 2 years under an F4E contract, the Tokamak Services for Diagnostics group at Wigner RCP has performed several experiments on MI cables, in close collaboration with F4E and IO team members. This paper outlines one of these experiments to measure the thermal gradient induced voltage in the cable cores. This was needed to evaluate the electrical noise levels due to thermal gradients in order to know if the noise has significant effect on the signal of some sensitive diagnostics. The subjects of the tests was a couple of Twisted Twin Mineral Insulated Cables, as this type of cables are used by the magnetics in the ITER machine.
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Neutronics pre-analysis and the status of neutron spectrum unfolding for the development of VERDI

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In future fusion power plants, such as DEMO, D-T neutron emission is predicted to exceed 1e21 neutrons/second. Accurately monitoring neutron energies and intensities will be the primary method for estimating fusion power, and calculating key parameters, including the tritium breeding ratio and nuclear heating. The Novel Neutron Detector for Fusion (VERDI) project, implemented under the EUROFusion Enabling Research 2017 program, aims to develop a novel neutron detector, capable of withstanding the harsh environment of a future fusion power plant. The VERDI detector is based on the foil activation technique, which relies neutron spectrum unfolding methods to process the convolution of gamma-ray measurement and detector response function and infer the neutron energy spectrum. A benchmark experiment has already been performed at the Frascati Neutron Generator, at ENEA, details of this experiment are provided in another submission. In this paper, experimental results from the Frascati Neutron Generator have been applied to neutron spectrum unfolding techniques. Calculations performed using MCNP and FISPACT-II were successful in predicting the gamma-spectroscopy peaks, and the initial use of MAXED has provided a neutron energy spectrum in good agreement with expectations. Results from the benchmark experiment have shown that careful treatment of (n,gamma) reactions are needed for neutron spectrum unfolding to obtain physically realistic solutions at low energy. The application of the VERDI detector is now being extended to JET to explore behaviour in both short-term and long-term experiments. This paper will focus on the pre-analysis calculations that have been performed in preparation for upcoming experiments during the JET D-D campaign. The status of neutron spectrum unfolding will also be discussed in this overview, along with plans for the development of neutron spectrum unfolding.

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Upgrade of Thomson scattering system on VEST

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Upgrade of the Thomson scattering (TS) system in Versatile Experiment Spherical Torus (VEST) is planned for measuring the electron temperature and density with higher reliability and higher time resolution. The existing TS system has difficulties on measuring single plasma discharge, since it uses a laser with energy of 0.65 J and repetition rate of 10 Hz, while the pulse duration of the plasma is ~20 ms. Recently, additional heating sources such as neutral beam injection and electron cyclotron heating are installed, thus, the upgrade of the TS system has been required to provide the time evolutions of the electron temperature and density for heating efficiency analysis. The upgrade is mainly related with two parts. First, a new laser is utilized for increasing both the signal-to-noise ratio (SNR) and the time resolution. The laser generates 10 pulses with 1 ms time interval in every 2 s, and each pulse has the energy of 2 J. And a switched capacitor type digitizer with 32 channels and 5 GS/s is newly employed for storing a number of pulse signals. Second, the optical fibers for transferring the collected TS photons are improved as well as the laser. Although the collection solid angle and the numerical aperture between the collection lens and the optical fiber are well matched each other, it uses only half of the maximum etendue of the polychromator. Therefore, the fiber bundle is designed to optimize the optical properties among the lens, fiber, and the polychromator. Compared to the current system, replacement of the laser, modification of laser injection optics, and fiber bundles are expected to improve the efficiency more than 6 times. Furthermore, in the temporal point of view, it decreases the number of required plasma discharges to be a tenth.

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Analyses of the influence of the recycling coefficient on He confinement in DEMO reactor

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Helium, as the ash of burning D-T plasma, is an unavoidable impurity component in DEMO reactor . Its efficient removal from the burning zone of a D-T fusion reactor is most important in the path towards achievement of economic fusion power production. Edge plasma transport properties: recycling/pumping will play a key role in the problem of helium removal from reactor.
This work describes integrated numerical modelling applied to DEMO discharges with tungsten wall and divertor, using the COREDIV code, which self-consistently solves 1D radial transport equations of plasma and impurities in the core region and 2D multi-fluid transport in the SOL. The model is self-consistent with respect to both the effects of impurities on the α-power level and the interaction between seeded (Ar) and intrinsic impurities (tungsten, helium). The coupling between the core and the SOL is made by imposing continuity of energy and particle fluxes as well as of particle densities and temperatures at the separatrix. In order to keep the prescribed plasma density at the separatrix, the deuterium recycling coefficient was iterated accordingly. It should be underlined that the recycling coefficient in our approach includes effects related to the pumping efficiency (albedo) as well as the intensity of the puffing.
The aim of the this work is to analyze the influence of the helium and hydrogen recycling on the He confinement in DEMO reactor. Simulation have been done for two argon seeding puff levels : moderate and strong.
It is found that recycling coefficient of helium have strong influence on the He confinement, which increases from 8.5s to 16s (going from lowest to highest recycling coefficient), but it has small influence on the effective charge state and radiation in SOL. Alpha power decreases only by about 10%, which is the effect of main plasma dilution.

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Feasibility of fusion fuel isotope detection below 1% using Penning gauge optical spectroscopy

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The species-selective (or Optical) Penning gauge approach to the measurement H2/D2/T2 fuel isotopic composition [1] and He/D2 concentration [2] in the neutralized particle exhaust of fusion devices is almost universally used nowadays across all fusion facilities. Although recent studies have shown that, through spectroscopic detection optimization, He/D2 detection is feasible down to at least 0.1% [3], an ongoing overview of existing data (JET) points to an apparent isotopic concentration detection limit of ~1% [4]. A laboratory study suggested that surface interactions with the isotopic species may be playing a role in this apparent limit [5].
In this paper, results from laboratory study, specifically designed to explore the impact of surface interactions, inside the Penning gauge, on this detectability limit will be presented. The study included baking of the gauge and a series of isotopic exchanges.
Penning bake-out at ~150℃ is found to dramatically reduce the H2 background in D2 only fueling cases. However, some self-cleaning by the Penning plasma discharges, after operating at high neutral pressure, can also be achieved.
A preliminary wall model for the gauge is used to interpret the findings of the laboratory study. The model presently includes hydrogen isotope retention in the surface, isotopic exchange and desorption from the surface.
Conclusions also include recommended procedures for optimization of isotopic composition analysis. The immediate application will be for the recently upgraded, divertor gas analysis system on JET

[3]. However, this will extend to other current devices, as well as to the ITER Diagnostic (DRGA) [4] .

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The development of technology of Be/CuCrZr joining using induction brazing

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Beryllium was selected as the plasma facing material for the ITER First Wall. Realization of the advantages of beryllium as a plasma facing material depends on the reliability of the critical beryllium joint with the heat sink made from CuCrZr alloy. This paper considers the method of induction brazing as the technology for this critical joint.
To prevent the formation of brittle intermetallics and preservation of the material properties of the CuCrZr alloy, it leads to the need to minimize the brazing temperature and time of the brazing cycle. This paper presents the result of selection of the optimal eutectic of STEMET 1101 brazing alloy thereby reducing the temperature of the brazing to 680 oC.
Using inductor allows to heat locally the beryllium tile/CuCrZr alloy region. The fast local heating of the braze region allows to melt the Stemet whilst ensuring that the major part of the component remains at a relatively ambient temperature. This is an important feature of the induction brazing technique as it will ensure the minimum time above the transition temperature for the CuCrZr and lead to acceptable material properties. The brazing is performed in a vacuum chamber to avoid contamination and oxidization of the component.
To fix each of the beryllium tiles, a unique clamping tool was developed and applied. This tool allows for an even distribution of the clamping force across the tile and ensures adequate clamping during the complete brazing cycle. The clamping tool does not overheat, can be quickly assembled, and is reusable
Along with ultrasonic testing of brazed joints, the most important criteria for the reliability of the beryllium armor is its behavior under cyclic heat fluxes. Authors present results of ultrasonic testing of the joints, as well as the first results of the high heat flux testing of the mock-ups.

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ITER magnetic sensor platform engineering analyses

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ITER in-vessel magnetic sensors play a key role for ITER plasma operation. Each of these sensors is accommodated in a platform mounted on the inner surface of ITER vacuum vessel and behind the blanket.
A full set of engineering analysis has been performed on the platform to assess the feasibility of the design configuration.
Electromagnetic (EM) Sub-Modelling technique has been used for very detailed evaluation of the EM loads die to plasma disruption events.
The EM sub-model has been built by taking one sample sensor with a portion of vacuum vessel around it and applying the vector potential obtained from ITER Global Model (IGM) analysis on the boundary nodes.
With this technique, platforms at 23 locations from a full poloidal array have been analyzed under 8 category III plasma disruption events. The two cases yielding highest EM load were selected to complete the structural assessment.
Thermal analysis was performed for a platform located on outboard midplane, where maximum heat load is expected.
The structural assessment has been performed as for the RCC-MR design code and considering a pessimistic input load combination of highest EM loads and highest thermal stresses over ITER operation lifetime.
An optimization process, dealing with EM and structural analyses, was set up, to reduce the EM loads and increase the structural strength of the platform.
The engineering analysis showed positive results on the structural integrity of the platform structure with the optimized design.
Future work is planned to incorporate the thermal deformation of VV surface due to nuclear heating and further refine the calculation.

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Implementation of Laser-induced Fluorescence Diagnostics in ITER

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A laser-induced fluorescence (LIF) diagnostic has been designed for measuring helium density (nHe) and ion temperature (Ti) in the outer leg of the ITER divertor. The LIF diagnostic is integrated with the divertor Thomson scattering (DTS) diagnostics via common injection and collection optics. Optimisation of previously proposed spectroscopic schemes, and lasers suitable for nHe and Ti measurements are the focus of this report.
The LIF method is based on laser pumping of transitions among excited states of atoms / ions and subsequent fluorescent signal processing. Optical parametric oscillators (OPO) or dye lasers with 5-10 ns pulse duration and repetition rate of ~ 1 kHz are suitable for nHe measurements. A set of spectroscopic schemes for He I detection requires the following wavelengths of laser radiation: 388.9, 501.7, 587.6 and / or 667.8 nm. Spectral coverage of OPO and dye lasers combined with currently available power spectral densities allow to achieve a saturation threshold in all the spectroscopic schemes under consideration.
Using time-modulated lasers (P ~ 20-50 W) for the Ti measurement is discussed in spectroscopic schemes with quenching excited states via pumping to higher excited levels. A tunable ytterbium fiber laser with 1012.3 nm wavelength can be used to quench the 468.6 nm line of He II. Spectral line scanning gives Ti from the absorption line contour Doppler broadening.
Optimization of the spectroscopic schemes and estimation of the required laser parameters have been performed using dynamic collisional-radiative models (CRMs). The schemes chosen were selected via comparing differences in measurement accuracy. The laser power spectral density has been estimated by calculating saturation threshold of the selected transitions in the ITER divertor plasma.

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Towards tritium measurements in W based on ps-LIBS diagnostics

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Determining tritium concentration within plasma facing components (PFC) of a thermonuclear reactor is crucial in terms of safety. As an example, tritium implantation can be high at the material surface (10-2 % at in W) and low in the bulk of the PFC (10s of ppm). In addition, a simultaneous implantation of tritium, deuterium and helium takes place. An in situ technique used to measure concentrations must enable isotopic discrimination and the access to T/D/He concentration profiles. In these conditions, Laser-Induced Breakdown Spectroscopy (LIBS) seems to be an appropriate method. LIBS is based on the interaction of a high energy laser pulse and the material. The irradiated material is ablated and transformed into low temperature thermal plasma whose spectroscopic analysis provides its composition and therefore that of the material.
LIBS using nanosecond pulses has been proved to be irrelevant [Mercadier et al, Journal of Nuclear Materials 414 (2011) 485] in determining the D/T ratio in metallic samples, a specific experimental setup has been elaborated based on the implementation of picosecond pulses. Results obtained on samples implanted with hydrogen isotopes will be presented. The depth of craters following the ablation has been measured. With pulse energy of ~ 20 mJ, low ablation rates (~ 100 nm/pulse) are obtained. The plasma spectroscopic analysis is performed on the [200, 800] nm spectral range for W (for fusion), Al (as Be substitute) and Si (as benchmark) implanted with mainly D+ ions produced by plasma discharges or ion beams. Temporal studies have been performed to identify the appropriate time window compatible with a plasma in thermochemical equilibrium. We will finally present concentration profiles obtained with a double pulse technique, when a second (5 ns) laser pulse is absorbed by the plasma to increase the signal to noise ratio and enables the measurement of He.

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Deformation and fracture behavior of the ODS-Cu/W joint fabricated by the improved brazing technique

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In our previous work, the joint between oxide dispersion strengthened copper alloy (ODS-Cu) and tungsten (W) demonstrated superior fracture strength (~200 MPa). In the present study, deformation and fracture behavior of the bonding layer and its vicinity after the three-point bending test was investigated. Consequently, it was found that the crack initiation site was dominantly in the tungsten bulk side, although it was not clear that the crack initiated from grain boundary or not. In addition, the crack propagation proceeded towards the bonding layer and seemed to end at the interfaces. These results indicate that the strength of the bonding interface is higher than that of the tungsten bulk and the present bonding technique is applicable for severe environments such as a high heat flux component on the fusion reactor.

The copper alloy has been considered to be useful as a divertor heat sink or cooling tube not only in the helical reactor FFHR-d1 but also in the tokamak DEMO reactor. The special feature of the basic option of the FFHR-d1 divertor is that the ODS-Cu, GlidCop® (Cu-0.3wt%Al2O3) is applied for the heat sink, and the flat plate tungsten armour is supposed to be bonded on the heat sink by using the improved brazing technique with BNi-6 (Ni-11%P) filler material.
The thickness of the bonding layer between GlidCop® and W was extremely narrow, and the joint strength and toughness were superior. This fact demands attention because it could be generally predicted that the bonding layer itself has a brittle feature. If we would want to obtain much higher fracture strength, tungsten bulk with better mechanical properties should be used. In this paper, the mechanism for producing such a fine and toughened bonding layer, and deformation and fracture behavior of the joint will be discussed.

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ITER Upper Visible/Infrared Wide Angle Viewing System: I&C design and prototyping status

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The ITER Upper Visible/Infrared Wide Angle Viewing System diagnostic will provide key measurements for machine protection and plasma control. The system, installed in five upper port plugs, will monitor the ITER divertor but also part of the ITER first wall using both high definition infrared and visible cameras typically running at 100 Hz. The plant system I&C will process about 40 Gb/s of imaging data acquired by the cameras installed in the port-cells up to 200 m away. Then surface temperature of plasma facing components will be computed and abnormal event such as hot spots or ELMs will be detected using dedicated algorithms implemented on fast controllers. Relevant data will be sent to plasma control system and to ITER archiving system for further offline analysis.
This papers focus on the I&C design for camera control, data acquisition electronics and analysis software. The current progress on I&C design following the ITER methodology for plant system I&C design is presented, including operational procedures, use cases, user and functional requirement tracing, functional analysis and plant I&C architecture. All design documents have been prepared using the Enterprise Architect software with dedicated plug-ins provided by the IO.
In parallel, prototyping activities related to real-time image processing and high throughput camera data acquisition using a 10 GigE interface have been carried out for the preliminary design review. The goal is to demonstrate some key functions of the system using state-of-the-art techniques adapted to the specific context and needs of the diagnostic. Emphasis is put on current challenges related to real-time computation and archiving of large amount of data.

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Thermometric chains for ITER superconductive magnets

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High environmental constraints are applied on the ITER magnets and therefore on their cryogenics thermometric chains. Accurate and reliable temperature measurements of ITER magnets and their

cooling circuits is of fundamental importance to make sure they operate under well controlled and reliable conditions. Therefore, thermometric chains shall reach a high operation reliability. In this paper, we present the full thermometric chain installed on the ITER magnets and their helium piping as well as the associated components production.
The thermometric chain is described from the sensor and its on-pipe assembly, to the signals conditioning electronics. The thermometric block design is based on the CERN's developed one for the LHC, which has been further optimized thanks to thermal simulations carried out by CEA to reach high quality level of industrial production. The ITER specifications are challenging in terms of accuracy and call for severe environmental constraints, in particular regarding irradiation level, electromagnetic immunity and distance between the sensors and the electronic measuring system. A focus will be made on this system, which has been recently developed by CEA: based on a lock-in measurement and amplification of small signals, and providing web interface and software to monitor and record temperatures. This measuring device provides a reliable and fast system (up to 100 Hz bandwidth) for resistive temperature sensors between a few ohms to 100 kohms.
During last two years, around 2500 thermo-blocks and nearly 50 crates of 5 boards, with eight channels each, have been produced and tested at CEA low temperature laboratory. All these developments and tests have been carried out thanks to three test benches built up at CEA Grenoble and in one industrial electronics laboratory. Test benches results on the entire production will be presented.
The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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Type tests of the ITER Switching Network Unit components and Protective Make Switches

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In ITER, each circuits of the central solenoid as well as poloidal field coils PF1 and PF6 is provided with a system for plasma initiation, called the Switching Network Unit (SNU), able to provide up to 8.5 kV for the coils. This will be obtained by inserting resistors in series with the pre-energized coils with the help of a DC current commutation unit (CCU) composed of connected-in-parallel mechanical bypass switch and a thyristor circuit breaker.
The paper describes the results of the work on development and testing of the main mechanical DC switches forming part of the CCU, namely, Bypass Open Switch (BPOS) and Bypass Make Switch (BPMS). These switches provide commutation of currents up to 55 kA at a voltage of 10 kV within 2-4 ms.
In addition, protective make switches (PMS) similar to the BPMS are considered. These devices are intended to shunt the AC/DC converters supplying power to the coils in case of their failure or in case of a quench, when the energy stored in the magnet system is extracted by a fast discharge to resistors.
After a short description of the design, the paper focuses on the procedure and results of the type tests on full-scale prototypes. In addition to electrical, hydraulic and pneumatic tests the test program implemented in 2017-2018 included EMC tests, functional tests with control system at rated currents and peak current tests at a current higher than 350 kA.
The successful results of the type tests confirmed the suitability of the design and compliance with the ITER requirements making it possible to start manufacturing of the switches for delivery to the ITER Site.

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Design Criteria of the Electrical Power Supply for Lithium Loop System of DEMO-Oriented NEutron Source (DONES) plant.

Co-authors: Pietro Zito [1]; Fabio Starace [1]; Mauro Cappelli [1]; francesco saverio nitti [1]

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The aim of this paper is to identify the design criteria of the Electrical Power Supply of the Lithium Systems of DEMO-Oriented NEutron Source (DONES) power plant. This facility is planned as a simplified IFMIF-like plant to provide, in a reduced time scale and with a reduced cost, information on materials damage due to neutron irradiation. In particular, a general overview of the current status of the electrical power systems developed for IFMIF-EVEDA has been made and also emphasis has been put on the analysis of the new specific requirements of DONES. This paper describes the Electric Power System (EPS) for Lithium Loop System (LLS). Its scope is to define design criteria, mainly focusing on the design of electric distribution system for electric loads of LLS and its components. A detailed description of design criteria and of key functions are reported, taking into account the specific requirements of the DONES Facility.
The Lithium Power System, has the following main functions:
 receives the AC power from the commercial power grid and transformed to proper voltage levels and feeds LLS electric loads in normal conditions;
 receives the DC/AC power from the uninterruptible power supply systems and emergency generator in case of the AC power grid lost, for SIC LLS electric loads;
 provides control and protection to the electric equipment, cables and electric loads against faults.
The Electric Distribution Systems provides the electric power to all electric loads of LLS. These loads are classified based on Safety Important Class (SIC) methodology into three groups based upon the loads served (in according to the IEC Standards):
• SR/Non-SIC loads, Single Power System (Normal Operation) fed by power grid;
• SIC-2 loads, Power System plus Emergency Power Supply System;
• SIC-1a/b loads, Redundancy Power System plus Emergency Power System.

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Paschen testing of ITER Central Solenoid qualification module

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General Atomics is currently fabricating superconducting magnet modules for ITER Central Solenoid in its Poway, CA facility. A critical step during final testing of the modules is high voltage checks of the insulation in Paschen conditions. A qualification coil was fabricated using the same techniques and equipment as the CS Modules. The qualification coil insulation was tested at voltages up to 30kV at pressures from 10-3 mbar to atmosphere to validate the CS Module insulation design and Paschen testing equipment.
Using the same vacuum chamber for cooling the coil to 4.5K, Paschen testing was performed utilizing a system to localize discharges without venting the vacuum chamber. Discharge events were detected using a high voltage tester and specially designed in-chamber multi-camera system.
The initial protection of the ends of the instrumentation wires failed and modifications were made to prevent Paschen discharges at the coil instrumentation wire ends.
Further testing revealed breakdown in the hand-wrapped area of the piping over the helium pipe weld where the instrumentation wires exited the ground insulation. Two modes of failure were identified: failure of the wire insulation (cracking) and failure of the ground insulation. The insulation scheme was redesigned and requalified. The final liquid helium pipe joint insulation scheme consisted of Glass-Kapton® (GK) tape wrapped with epoxy resin. Methodologies were developed to fill "cusps" between round wires and round pipes and to eliminate a problem of wire insulation cracking after resin curing. This insulation scheme was implemented on the qualification coil, which then passed 30kV in all prescribed Paschen conditions.

This paper describes the qualification coil Paschen test results, development of Paschenization techniques to be used on ITER CS Modules, development of the insulation joints, and method of extracting instrumentation wires on the helium supply and return pipes while maintaining high voltage standoff in Paschen conditions.

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Design and simulation of a cascaded four-quadrant 24- pulse converter based on 6-phase pulsed motor-generator

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This paper designs a cascaded four-quadrant 24-pulse converter fed by a 6-phase pulsed motor-generator (M-G) aiming at the requirements of HL-2M tokamak, which is mainly used for the control of vertical instability of plasma. The optimally designed four-quadrant 24-pulse converter cascaded by four-quadrant converters is able to balance the loads of the double Y of M-G. Owing to the fact that the control of the four-quadrant converters are relatively independent and logical controlled with circulating current, it is available to work under high current in four-quadrant. The output current transits smoothly and continuously over the zero crossing point. In addition, the circular reactors can be saved by taking full advantage of the large inductance of the M-G to restrain circulating current. This paper designs a four-quadrant 24-pulse converter satisfied with the requirements and makes MATLAB simulations to verify the performances of current following, the adjustability of voltage source fluctuating, the smoothness of current passing zero and the presentation of circulating current.

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Repair processes of W7-X target modules

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The highly loaded surface of the actively water-cooled divertor of Wendelstein 7-X (W7-X) is made of 100 individual target modules. In each target module, a set of target elements is water-cooled in parallel and fed by manifolds. A target element is made of a CuCrZr copper alloy heat sink, armored with CFC NB31 tiles. Due to the width of the target elements, CFC tiles had to be successively electron beam welded onto the heat sink from two sides.
He leak testing under pressure in a vacuum oven was systematically performed for each target element and module. The type 5S target element had a higher percentage of rejection rate during the production, and one series element did not pass the leak test after high heat flux testing. The leakage was the result of the combination of a porosity concentration due to the CFC-CuCrZr - weld in the centre part of the element and the reduced CuCrZr thickness due to machined slots at this location. The selected solution was sealing of the slit between two tiles by electron beam welding.
The target elements were then connected by orbital welding to the piping system. One of the first produced batch of 30 target modules did not pass the integral He leak testing. Two leaks were detected in the weld seam between two target element connectors and manifold pipes. Their positions did not allow a reliable process for re-welding from outside due to the impossibility to install any jigs. The selected approach was drilling of apertures through the neighbouring manifolds to allow direct access to the leaking seams from inside. The openings allowed installation of an inside orbital

welding head. This solution was validated in an intensive prototyping phase. Finally, the repaired target modules passed the He leak test in oven.

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A European design proposal on the ITER ELM-Coil Power Supply optimized for ELM mitigation and RWM stabilization

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The paper describes a design proposal for the ITER ELM Coil Power Supply, optimized for the simultaneous ELM mitigation and RWM stabilisation during the ITER non-inductive operation. Slow (ELM) and fast (RWM) rotating magnetic fields are generated by exciting the three sets of nine ELM-Coils at ELM-frequencies up to 5 Hz (N = 4) and RWM-frequencies up to 60 Hz (N = 1). Starting from a basic concept of the power supply scheme proposed by IO, the EU-DA developed, studied and optimized the power supply and control systems, by means of extensive computer models. By combining the nine DC-AC IGBT Inverter Power Units on the same DC-bus, the AC-DC converter is optimized but substantial current harmonics at frwm ± felm are generated in the DC-system. In turn, these excite low frequency LC-resonance arising from the distributed DC-capacitor banks and stray inductance of DC-power connections. Further cancellation of the current harmonics at the DC-system level is primordial (impact on the grid) and requires an accurate phase control of the nine ELM-Coil currents. The optimized power and control scheme, featuring commercial compact water-cooled IGBT Inverter Assemblies with own DC-bus, achieves very good performance in normal operation (amplitude error < 1%, phase error < 1°), is shown to be immune to power system imbalance and is well protected against plasma disruption through the synchronized triggering of bi-polar thyristor crowbars. The equipment layout and component ratings of the power supply scheme have been completed. In view of the limited space at the level-4 of the Tokamak building, a compact layout was developed, in which the AC-DC Converter and nine DC-AC IGBT Inverter Power Units are assembled on a two-level metallic structure, designed to be lifted complete by the ITER station crane.

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Fabrication Status of ITER Centrol Solenoid Modules

Co-authors: John Smith [1]; Nikolai Norausky [1]; Donna Priddie [1]; Kurt Schaubel [1]; Alan Stephens [1]

[1] *General Atomics*

The fabrication of the modules for the ITER Central Solenoid (CS) is in progress at General Atomics (GA) in Poway, California, USA. This purpose built facility has been established with the requisite tools and machines to fabricate the seven 110 tonne CS modules (six required plus one spare). The current schedule has the first module's fabrication completing in 2018 followed by electrical and full current testing. GA has completed the fabrication of a qualification coil, simulating all of the processes and exercising all of the tooling required to fabricate a production module. After being completed, the qualification coil underwent a cool down cycle to 4.5K which served as the final commissioning step of the helium cooling system on the parallel flow paths of a module. In addition, other final test station systems and procedures were simulated to verify their functionality including simulated Critical Current Sharing Temperature measurements, DC switch operation to dump the energy from the coil, and a helium leak test on the coil at <10K. After warming the coil, Paschen tests were repeated prior to the coil being dissected to evaluate the quality of the vacuum pressure impregnation (VPI). Dissection is scheduled for early spring 2018.

GA currently has four production modules in fabrication, all in different stages of production. The first module will be vacuum pressure impregnated in the spring of 2018. The second module is currently in the insulation process with the third module being joined together into a 110 tonne module. Module 4 is wound. The description of the VPI process planned for late spring 2018 will be described. To verify the turn insulation of a module, unique impulse testing methods were developed to test the insulation between the 560 turns of a module and these methods are reported in this paper.

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Lithium Loop and Purification System of DONES: Preliminary Design.

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The Demo-Oriented NEutron Source (DONES) is an essential irradiation facility for testing candidate materials for DEMO reactor and future fusion power plants. An intense flux of highly energetic neutrons is generated by the nuclear reactions of a 125mA beam of deuterons at 40MeV striking a liquid lithium target. The neutron flux achieves a damage rate of 8-10 dpa/fpy in a volume of about 0.3 l with a helium production rate of ~10-13 appm He/dpa. The main lithium loop and the related purification system have to generate a stable lithium jet at the target and guarantee: a high speed flow to evacuate the deposited heat (5 MW) and avoid boiling or significant evaporation of the lithium; a constant shape and thickness of the jet to assure a constant neutrons flux and prevent impingement of the beam on the back-plate, being the latter the surface just behind the jet; and an adequate level of chemical impurities solved in lithium. The preliminary design of the two systems is concluded. In this work, the lay-out of the loops, piping dimensioning, pressure drop evaluations, definitions of supports, piping stress analysis, and the design of the traps to remove the impurities are described in detail.

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Fault analysis and overvoltage estimation in the DEMO Toroidal Field coil circuit

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In the European DEMOnstration nuclear fusion power plant (DEMO), the desired toroidal magnetic field is produced by a magnet system composed of 16 Toroidal Field (TF) coils, according to the last 2017 reference baseline. The total stored energy of about 140 GJ, more than three times that of ITER TF coils, has to be quickly dissipated in case of quench by a suitable Quench Protection (QP) system. The energy, the current and the required discharge time constant define the voltage to be applied to the coils; however, the peak value at the coil terminals during the fast transient phase at the beginning of the discharge or in case of faults can be much higher.

This paper deals with first studies addressed to estimate maximum voltage stresses for TF coils in various operating conditions and for different TF circuit topologies to evaluate their relative merit and to provide inputs for the definition of the number of toroidal sectors as a compromise between requirements for the coil insulation and cost and size of protection system, busbars and current leads. The studies were firstly done with 18 TF coils (2015 reference baseline); since three different winding pack options are under study, we selected the one characterized by the highest inductance, thus by the related highest voltage across the coils at the discharge.
Numerical simulations were carried out to reproduce the voltage waveforms at the coil terminals, across the coils and to calculate the i2t in the coils for the cases analyzed for the different TF QP circuit topologies, operating conditions, and number of sectors (18 / 9). Then, the analyses have been updated for the case of 16 / 8 sectors of the last reference baseline; all the main results are reported and discussed.

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Examination of ITER Central Solenoid prototype joints

Co-authors: Stefano Sgobba [1]; Sandra Castillo Rivero [1]; Pilar Fernandez Pison [1]; Paul Libeyre [2]; David Everitt [3]

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The ITER Magnet System will be the largest and most challenging integrated superconducting magnet system ever built. For the Central Solenoid (CS), cable – in – conduit - conductors (CICCs) of nearly one kilometre length are produced, but still, it will be necessary to connect several lengths together to wind the gigantic 110 tonnes coils. The creation of these superconducting joints is one of the most delicate parts of the assembly.

There are three types of ITER CS joints: sintered joints, coaxial joints and twin – box joints. US ITER, the ITER Domestic Agency of the USA produced a prototype containing all three types of joints. The goal is to test the performance of the joints in the SULTAN facility of the Swiss Plasma Center, capable of reproducing close – to – service conditions: high magnetic field (up to 11 T background field), high current (up to 100 kA) and high mass flow rate of supercritical helium for cooling.

The paper describes the results of a comprehensive examination campaign aimed at understanding the relation between fabrication and performance of the CS prototype joints. The test campaign combines advanced image analysis for the assessment of the void fraction of the conductors and dimensional measurements performed at different cross-sections of the joints, scanning electron microscope (SEM) and energy dispersive x-ray spectroscopy (EDX) to evaluate the quality of the contact between the strands and the sleeve / sole, as well as a full assessment of the welds according to the most stringent acceptance levels of the standards in force.

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Temporary stagnation in the recrystallization of warm-rolled tungsten in the temperature range from 1150 °C to 1300 °C

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Pure tungsten is a potential candidate for armor material of fusion reactors as it possesses superior thermal properties and radiation resistance. Application at the desired operation temperatures for longer times will result in a loss of strength accompanied by embrittlement due to thermal

activated changes in the microstructure, in particular due to recrystallization, undermining tungstens outstanding performance. Investigating the thermal stability of tungsten depending on the manufacturing process is therefore considered crucial. The thermal response of a sintered, hot isostatic pressed tungsten plate warm-rolled to 80% thickness reduction is assessed in the temperature range from 1150 ℃ to 1300 ℃. The restoration processes occurring during annealing are tracked by changes in mechanical properties through hardness testing and identified by supplementary orientation mapping by means of EBSD. Isothermal annealing treatments were performed at six different temperatures. With increasing annealing time, the macro hardness decreased continuously; different stages corresponding to different stages of the microstructural evolution (recovery, recrystallization and grain growth) could be identified and confirmed by the microstructural information gained by EBSD. For the time to half recrystallization, an activation energy comparable to the activation energy of bulk self-diffusion is obtained. The correspondingly extrapolated times to half recrystallization would not allow to operate the material at temperatures above 1100 ℃. Nevertheless, for all annealing temperatures a stagnation period in the evolution of the macro hardness was observed where the hardness loss and hence the degradation of mechanical properties halted for a significant amount of time, before it resumed. EBSD investigations revealed that the stagnation occurred when tungsten was still only partially recrystallized, and that recrystallization recommences afterwards. Such a temporary resistance against complete recrystallization and revealing its microstructural origin is of uttermost importance as its understanding could provide interesting insight in opportunities for designing tungsten material with improved thermal stability.

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Conceptual Design of a Toroidal Field Coil using HTS CrossConductor

Co-authors: Reinhard Heller [1]; Walter Fietz [2]; Thomas Vogel [1]; Klaus-Peter Weiss [1]; Michael J Wolf [1]

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The potential of High Temperature Superconductor (HTS) for a Toroidal Field Coil (TFC) of a future fusion power plant has already be demonstrated in a conceptual design within EUROfusion [1]. One of the candidates of a high current HTS conductor for use in a fusion magnet is the so-called HTS CrossConductor (HTS CroCo) where REBCO tapes are arranged in a cross sectional optimized way. The basic TFC dimensions have been taken from the PROCESS system code as the starting point for the design of a winding pack, consisting of six CroCos around a copper core, embedded in a stainless steel jacket and cooled by 4.5 K supercritical helium. With this cable geometry, the electromagnetic, structural mechanics, cooling, and thermo-hydraulic performance of an HTS-TFC were investigated. It could be shown that the conductor and winding pack design fulfills the requirements with respect to structure mechanics and hot spot in case of a quench. The current sharing temperature is large enough that it is possible to handle the nuclear heat load on the coil with still sufficient margin.
[1] R. Heller, P. V. Gade, W. H. Fietz, T. Vogel, K. P. Weiss, "Conceptual Design Improvement of a Toroidal Field Coil for EU DEMO using High Temperature Superconductors", IEEE Trans. Appl. Supercond. 26(4) (2016) 4201105

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The ITER in vessel coils – design finalization and challenges

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A design update of the ITER In-Vessel Coils (IVCs) has been launched after the prototype coil manufacture in 2014 revealed some major issues in particular related to brazing and joints inside the coils. In parallel a review and update of the plasma operating scenarios and requirements of the IVCs system has been done and a refined set of plasma pulses and corresponding load scenarios of the IVCs has been elaborated in collaboration with the ITER plasma control team. For the fatigue analysis of the vertical stabilisation (VS) coils, the spectrum type electromagnetic loading has been analysed by applying the Rainflow cycle counting method. Furthermore, the maximum currents during transient plasma events have been assessed considering actual operating currents leading to more representative load cases. The main IVC component modifications are the conductor material and the winding pack support structure. In view of risk reduction and cost efficiency, the conductor design has been unified among the IVC system. Cu brazing has been replaced by welding and the conductors are not brazed to the winding pack bracket but clamped. Huge effort has been made in the structural analysis and design integration of the IVC components. With a strong support from the design integration and teams of the other ITER in vessel components, a fully integrated design baseline of the IVC components has been established.
Feasibility studies and mock-ups performed in the frame of the design finalization have identified the main critical activities during manufacture such as the joint assembly, the ELM coil bracket welding and the in-situ winding of the VS coils. The status of the IVC conductor procurement involving two suppliers for the first phase covering process qualification is being shown as well. Key challenges experienced during this first phase along with results on mechanical and electrical tests are presented.

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Effect of coil configuration parameters on the mechanical behavior of the superconducting magnet system in the helical fusion reactor FFHR

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FFHR depicts the conceptual design of an LHD-type helical fusion reactor that is being developed by the National Institute for Fusion Science. Several design mechanisms for FFHR have been investigated. For instance, FFHR-d1A is a self-ignition demonstration reactor that operates at a magnetic field intensity of 4.7 T and has a major radius of 15.6 m. FFHR-c1 is a compact-type sub-ignition reactor that intends to achieve steady electrical self-sufficiency and that depicts a high magnetic field intensity of 7.3 T and a small major radius of 10.92 m. For both types of FFHR, the shape and positional relation among the superconducting coils, a pair of helical coils with two sets of vertical field coils, are observed to be similar with each other. Such a relation is based on the coil configuration of LHD. The coil configuration is defined using an aspect ratio, a pitch parameter of the helical coil, the number and geometric position of the vertical field coil, and so on. There is an increasing demand to achieve an optimized coil configuration to anticipate the improvement in plasma-confinement conditions. However, there are a few investigations that study the effect of coil configuration on mechanical behaviors of the coil systems, including the support structure. In this study, the structural design of FFHR-c1 based on the fundamental set of parameters of coil configuration is depicted, which satisfies the soundness of the structure. Further, the effects of the coil configuration parameters on the stress/strain distributions are investigated. A multiscale analysis is performed simultaneously to facilitate a detailed investigation of the elements in the superconducting coil. Furthermore, the mechanical behaviors during the coil excitation process are evaluated based on the contact between the coil winding and the support structure.

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Maximization of the magnetic flux generated by a DEMO CS coil using HTS conductors

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The present work is performed within the framework of the EUROfusion DEMO project. Previously, it was demonstrated that for a maintained magnetic flux the use of HTS conductors at highest magnetic field in a layer-wound CS coil would allow the reduction of its outer radius by around 0.5 m as compared to the DEMO reference design using only Nb3Sn conductors. Alternatively, the superior high field performance of HTS conductors can be used to maximize the magnetic flux of a CS coil with a given outer diameter, which could significantly prolong the duration of plasma burn phase and thus the overall power plant efficiency. In the present study, the maximization of the magnetic flux, generated by a layer-wound CS coil of fixed outer radius, is considered. HTS conductors are envisaged to be used at highest field, while Nb3Sn and NbTi are foreseen to be used in intermediate and low field layers, respectively. The inner radius of the CS coil is optimized with respect to the generation of a maximum flux taking into consideration the superconductor properties, the hoop stress and the axial stress. In order to provide reasonable starting values for the finite element analysis, first a simplified model with layer-dependent current densities, however, without stainless steel grading will be considered. In the final outline design of the CS coil a superconductor and stainless steel grading will be implemented.

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Joints for cable-in-conduit conductors

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A review of the joints for the ITER CS CICC is given. More detailed discussion of the design and performance of the ITER CS joints is presented including successes and the revealed problems. ITER CS has three types of joints: 1) sintered joints to connect conductor lengths in the CS module; 2) coaxial joints to connect the CS module terminations to the superconducting buses; 3) twin box joints, which are bi-metal copper -stainless steel boxes containing the compacted cables that connect the buses to the ITER feeders. All the ITER CS joints have different requirements for space allocation and ability to be assembled/disassembled. Thus, different designs of the joints are used. The CS joints went through the R&D and qualification effort, but recent verification results showed a necessity of making the design more robust by revisiting some issues that seemed to be established to satisfaction in the qualification phase. We discuss requirements for the strands treatment, quality of interfaces, heat treatment conditions and other technological issues. We also discuss the effect of oxidation of the cable to copper surface interface, effect of the joint compaction, Cr removal and soldering parameters on performance of the joint. Recent results of the ITER CS joints performance and autopsy studies are presented.

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Modelling of chemical vapor deposition to improve tungsten fiber reinforced tungsten composites (Wf/W)

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Due to the unique combination of excellent thermal properties, low sputter yield, hydrogen retention and activation, tungsten is the main candidate for the first wall material in future fusion devices. However, its intrinsic brittleness and its susceptibility to operational embrittlement is a major concern. To overcome this drawback, tungsten fiber reinforced tungsten composites featuring pseudo ductility have been developed. Bulk material can be successfully produced utilizing chemical vapor deposition of tungsten fabrics. However, a fully dense composite with a high fiber volume fraction is still a huge challenge.

Therefore, a model is currently developed in COMSOL including the complex coupling of transport phenomena and chemical reaction kinetics. To validate the model with experimental data, fibers were deposited in heated tubes under controlled parameter variation. The temperature and tungsten growth rate were measured along the fibers and inner tube surfaces for different heater temperatures, partial pressures and gas flows. With the experimental results the prediction of the model has been improved. As next step the model will be applied to design infiltration experiments to fabricate fully dense Wf/W composites with a high fiber volume fraction.

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Recent experiments with the European 1MW, 170GHz industrial CW and short-pulse gyrotrons for ITER

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The European Gyrotron Consortium (EGYC) is developing the EU 1 MW, 170 GHz CW industrial prototype gyrotron for ITER in cooperation with the industrial partner Thales Electron Devices (TED) and under the coordination of Fusion for Energy (F4E). This hollow cylindrical cavity gyrotron is based on the 1 MW, 170 GHz short-pulse (SP) modular gyrotron that has been designed and manufactured by KIT in collaboration with TED. The experiments with the CW industrial gyrotron are organized in two phases. The first phase was completed successfully at KIT in 2016. In the SP regime (<10 ms pulses), stable excitation of the nominal cavity mode TE32,9 at 170.22 GHz was achieved for a wide range of operating parameters. The maximum RF output power of the tube is higher than 0.9 MW with a total efficiency of 26% in non-depressed collector operation. The Gaussian mode content of the RF output beam is higher than 97%. In long-pulse operation, pulses with duration of 180 s (limited by the high-voltage power supply at KIT) delivered power higher than 0.8 MW with 38% efficiency (in depressed collector operation). The second phase of the experiments is ongoing

at SPC, Lausanne, with the goal to further optimize the output power and extend the pulse duration. In parallel, the experiments with the SP prototype are continued at KIT. The SP tube, which in multiple experimental campaigns delivered power higher than 1 MW with 42% efficiency (in depressed collector operation), is further upgraded. Various depressed collector operation schemes are tested with the goal to achieve an efficiency higher than 50%. Moreover, different beam tunnels will be tested in order to have the possibility to go to higher operating beam currents without exciting parasitic oscillations. In this work, the latest results with the CW and SP prototype gyrotrons will be presented.

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Hydrogen isotopes distribution modeling by "FC-FNS" code in fuel systems of fusion neutron source DEMO-FNS

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Tokamak-based fusion neutron source (FNS) [Kuteev B.V. et al 2010 Plasma Phys. Rep. 36 281, Kuteev B.V.et al Nucl. Fusion 55 (2015) 073035] is the centerpiece of the fusion-fission hybrid reactor (combining nuclear and thermonuclear technologies). In Russia, for the demonstration of stationary and hybrid technologies, the DEMO-FNS project has been developed, which should operate at least 5000 hours per year. The plasma operation in a tokamak requires the continuous injection of the fuel mixture containing hydrogen isotopes (deuterium and tritium) into the vacuum chamber, as well as its subsequent pumping out and processing. Calculation of the distribution of tritium (as well as deuterium and protium) in fuel systems is important for assessing safety features of the facility and for designing these systems. To simulate hydrogen isotope flows and inventories in the fuel systems of FNS, computer code FC-FNS [Ananyev S.S. et al Fusion Eng. Des. (2016), Volumes 109–111, Part A, pp 57–60] has been created that continues to be developed. This report describes capabilities of the code. The results of calculations are presented for the conceptual design of DEMO-FNS. The balance of three hydrogen isotopes is taken into account, the performance of deprotiation systems is calculated to maintain the required level of protium in the plasma tokamak and detritiation (for the variant of the neutral injection system - NBI). Three alternative scenarios for supplying gas to NBI system are simulated. It is shown that the proposed approach to NBI fueling allows reducing the total amount of tritium in FS up to 1.5 times, that leads to the initial loading for DEMO-FNS of 460 g. The time for tritium breeding up to the amount sufficient for starting a new similar device will be 2.5 - 4 years (for different scenarios for FS NBI).

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Overview of the JET Operation Reliability

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The JET tokamak has been in operation since 1983, producing ~92500 pulses so far. For the period 2000 to 2016 (not including DTE1 in 1997), information on every shutdown, commissioning phase and experimental campaign has been logged, providing unprecedented operation reliability statistics and a model for studying reliability, availability, maintainability and inspectability (RAMI) in fusion experiments such as ITER.
The JET Operation Reliability Statistics record the time taken to install major upgrades, commission systems with plasma, the downtime for maintenance and the achieved versus planned experimental campaign days. On average, JET achieves 85% of the planned experimental days and up to 180 campaign days during a calendar year.
For unplanned interventions the systems that failed are identified, the impact on machine availability

is logged and the remedial actions taken are documented. A total of 12 unplanned interventions were recorded over 17 years with durations in the range 2 to 8 weeks.
Delays during operations are recorded, providing a detailed overview of faults from essential subsystems such as computer systems, power supplies and neutral beam systems and delays attributed to human operators. On average 21% of the time is lost due to delays during experimental campaigns. This has been consistent over the last 17 years and is compensated by having 20-25% contingency for campaign planning.
Maintenance schedules and refurbishments are planned at JET, using the operation reliability data. Recently, pulsed power supplies and heating systems were refurbished, enabling continued high operational availability of JET for the upcoming campaigns using deuterium-tritium mixtures in 2019-2020.

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Optimization of RFX-mod2 gap configuration by estimating the magnetic error fields due to the passive structure currents

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The design of a major refurbishment of the toroidal complex of the RFX-mod experiment is going to be finalized before starting the realization phase. The Inconel vacuum vessel will be removed and the stainless steel supporting structure will be modified so as to become vacuum tight. The plasma facing graphite tiles will be mounted onto the inner surface of the copper shell, allowing an increase of the plasma proximity factor. This could allow different operation regimes expected to provide a significant reduction of the amplitude of RFP tearing modes. This reduction would lead to magnetic chaos mitigation with confinement improvement, better mode control capability and reduced plasma wall interaction. On the other hand, due to the shorter distance from the passive structures, the plasma will be even more sensitive to magnetic field errors produced by the induced currents at the cuts of the same structures. A careful assessment of different gap configuration is mandatory to optimize the final design of the machine modifications. Numerical analyses confirmed by the following experimental results showed that overlapping layers at the shell poloidal gap were an effective solution to minimize error fields in RFX-mod. This suggested analysing the possibility of a similar configuration at the shell inner equatorial gap, too. The direct reproduction is not straight-forward due to strict dimensional constraints and the required compatibility with a much more complex assembly procedure necessary to make the support structure vacuum tight. Parametric analyses of different constructive solutions have been made possible by a computational tool specialized to estimate the induced currents in thin conducting structures with complex geometry and the associated magnetic fields. The magnetic error fields due to other new features of the design such as the conducting cage introduced to assure the electrical equipotentiality of the graphite tiles have also been investigated.

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EUROfusion Plasma EXhaust (PEX) strategy on upgrades to European tokamaks and PFC test facilities

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The European fusion electricity roadmap sets out a strategy for a collaboration to achieve the goal of generating fusion electricity by 2050. It has been developed based on a goal-oriented approach

with eight different missions including development of Heat-Exhaust systems which must be capable of withstanding the large heat/particle fluxes of a fusion power plant. This paper summarises the development of the strategy to achieve this mission, focussing on the programme of coordinated upgrades to European tokamaks and PFC facilities.
The strategy pursues the conventional ITER divertor solution in a combination of radiative cooling and detachment, which is taken as the baseline for DEMO. Nevertheless, a significant risk remains that high-confinement regimes of operation are incompatible with the larger core radiation fraction required in DEMO. Therefore, an investment in assessment of the adequacy for DEMO and proof-of-principle tests of innovative geometries as well as the use of liquid metals (LM) is also required. The broad tasks and milestones for both the conventional and risk mitigation approaches for developing the solutions at the levels of Proof of Principle, Demonstration and Qualification, including implementation on DEMO will be summarised.
To enable implementation of this strategy a dedicated programme of PEX facilities upgrades and new devices was developed. The selection of projects in the strategy was based on the degree to which the projects enabled the physics and technology gaps to be addressed, prioritising of studies on conventional materials and divertor configurations. Currently the programme includes upgrades on the three MST tokamaks (AUG, TCV and MAST-U), WEST tokamak and a hot cell facility. The assessment PEX activities and the decision to down select alternatives for possible implementation on DTT are scheduled for 2023-2024. The results of this analysis and current status and progress of the PEX upgrades will be presented and discussed.

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Multiphysics Analysis of W7-X Control Coils

The quasi-symmetric fivefold modular Wendelstein 7-X (W7-X) stellarator consists of three groups of coil systems, i.e. superconducting magnet, trim coil and control coil systems. The control coil system contains ten identical 3D shaped control coils (CC) situated behind the baffle plates of corresponding divertor unit, and is designated to rectify the error field and to sweep hot spots on the divertor target plates. The CC is wound from copper conductor with a square cross section of 16 mm x 16 mm and a water cooling hollow of Ø 8 mm. The control coil system was installed in W7-X in 2015, and the integral commissioning has been done in parallel with the completion of W7-X. During the operation phase (OP 1.2a) with limited plasma heating power, a leakage in one of the CC cooling water plug-in was found and dictates a detailed transient thermal analysis of CC to determine the allowable operation time without cooling water flow. The paper presents the transient thermal analysis and is followed by a detailed finite element mechanical analysis with the consideration of temperature gradient loads, dead weight and electromagnetic forces. Moreover, the transient thermal and mechanical performance of actively cooled CC to be intensively operated during state steady operation phase (OP 2) are also analyzed and evaluated with the same FE model.

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On optimization of air cooling system of FDRs dissipating energy from ITER magnet coils

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The Fast Discharge Resistors (FDRs) under development at NIIEFA are intended together with switching equipment to dissipate energy released in case of a quench of the ITER superconducting coils, thereby protecting them against failure. FDRs are made of sections consisting of steel resistive elements enclosed in boxes. Two-four sections stacked vertically form a separate module. During

energy release the resistive elements are heated to 250-300^{0}C practically adiabatically. The resistors should be cooled to their initial temperature within 3-4 hours. For this purpose, the authors have developed the air cooling system based on the forced air circulation produced by seventeen fans in a complex system of series-parallel channels formed by air supply and return pipes, vertical modules and chimneys. The numerical simulation of the cooling process revealed that distribution of the air flow in the parallel channels formed by the vertical modules is considerably non-uniform, which essentially increases the module cooling time. The study performed by the authors within the last few years has made it possible to propose measures on optimizing the air cooling system mitigating the negative effect of air flow non-uniformity in the FDR modules. The reported analysis continues the previously performed studies of the FDR cooling system. The idea to install diaphragms in each module to equalize the air flow in the cooled-in-parallel modules has made it possible to reduce the time for their cooling to the specified values without a considerable reconfiguration of the air cooling system.

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Investigation of the thermal expansion of lithium orthosilicate

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Within the EU, the current grade of advanced ceramic breeder pebbles is composed of a mixture of Li4SiO4 (LOS) and Li2TiO3 (LMT). These pebbles are fabricated at KIT by the melt-based process "KALOS". The addition of LMT is beneficial for two aspects: the mechanical strength of the pebbles is considerably increased and the long-term stability at high temperatures is improved. Nevertheless, the advanced ceramic breeder pebbles generally show a number of defects that decrease the ideally achievable mechanical strength. In this work, high-temperature X-ray diffraction (HT-XRD) experiments will show that the well-known displacive phase transformation of LOS at temperatures between 665 ℃ and 723 ℃ is strongly anisotropic. The formation and growth of the defects of the pebbles may thus not be exclusively attributed to thermal stresses during the fabrication of the pebbles but also to this phase transformation. To address this issue, principally two ways exist. Either the intensity of the phase transformation is attenuated, or the phase transformation is shifted to unobjectionably low temperatures. In this study both ways as well as their combination are demonstrated. It will be shown that the solid solution of germanium in the LOS lattice is an effective way of attenuating the phase transformation while the partial substitution of lithium by magnesium shifts the phase transformation to lower temperatures. Furthermore, it is also demonstrated that by adding magnesium the thermal expansion coefficient of LOS can be aligned with that of LMT, so that intrinsic stresses are reduced as additional benefit. HT-XRD, dilatometry as well as differential scanning calorimetry are used for the analysis of different compositions to evaluate the individual effects of the added elements. Eventually, the impact on the pebble strength is demonstrated for selected promising compositions.

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Type tests of the ITER Switching Network Unit components and Protective Make Switches

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In ITER, each circuits of the central solenoid as well as poloidal field coils PF1 and PF6 is provided with a system for plasma initiation, called the Switching Network Unit (SNU), able to provide up to 8.5 kV for the coils. This will be obtained by inserting resistors in series with the pre-energized coils with the help of a DC current commutation unit (CCU) composed of connected-in-parallel mechanical bypass switch and a thyristor circuit breaker. The paper describes the results of the work on development and testing of the main mechanical DC switches forming part of the CCU, namely, Bypass Open Switch (BPOS) and Bypass Make Switch (BPMS). These switches provide commutation of currents up to 55 kA at a voltage of 10 kV within 2-4 ms. In addition, protective make switches (PMS) similar to the BPMS are considered. These devices are intended to shunt the AC/DC converters supplying power to the coils in case of their failure or in case of a quench, when the energy stored in the magnet system is extracted by a fast discharge to resistors. After a short description of the design, the paper focuses on the procedure and results of the type tests on full-scale prototypes. In addition to electrical, hydraulic and pneumatic tests the test program implemented in 2017-2018 included EMC tests, functional tests with control system at rated currents and peak current tests at a current higher than 350 kA. The successful results of the type tests confirmed the suitability of the design and compliance with the ITER requirements making it possible to start manufacturing of the switches for delivery to the ITER Site.

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Registering micro-indentation of neutron-irradiated low-activation steel at high temperatures

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An ongoing study about the influence of neutron irradiation on the mechanical properties of the first wall's structure materials is presented in this work. EUROFER97 and an Oxide Dispersion Strengthened EUROFER steel were irradiated in the Petten High Flux Reactor up to a nominal dose of 15 displacements per atom at temperatures between 250 and 450℃ and investigated by an advanced method of registering micro-indentation. For the purpose of characterizing irradiated and thus radioactive samples at future fusion reactor conditions i.e. at high experimental temperatures, the Karlsruhe High Temperature Indenter (KAHTI) was developed. In order to safely handle radioactive samples, KAHTI itself is operated by remote control inside a Hot Cell. Due to a new optical depth sensing method, it now is possible to perform registering micro-indentation at temperatures up to 650℃ with highly accurate displacement measurement. The results gained by KAHTI show the increase of hardness by neutron irradiation and its dependency on the irradiation temperature. In addition, the sensitivity of the hardness to different testing temperatures is investigated. These results lead to a better understanding and quantification of the different neutron induced damages. The possibility of recombining neutron induced Frenkel-pairs and thus the reduction of irradiation damage without affecting the material's grain structure is investigated by performing post irradiation annealing inside KAHTI at 550℃. First results indicate an almost complete recovery of mechanical properties. Comparing the results of this study to conventional testing methods approves KAHTI being a complementary testing method with a very high specific amount of information per sample and per sample volume.

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Integrated core-SOL-divertor modelling for DTT tokamak with liquid metal divertor targets

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The behaviour of the SOL plasma of the Italian projected DTT is analysed for the standard divertor configuration by means of the integrated COREDIV code simulations when either Lithium or Tin are used as liquid target materials.
The DTT tokamak is expected to operate in H-mode, which requires the value of power to scrape-off layer above the L-H threshold. On the other hand it is postulated that the divertor power load should be controlled to not exceed 5MW/m2. Cooling of the plasma can be achieved by seeding or by intrinsic impurities like particles released from the divertor. In the case of liquid metal divertor, vaporization additionally enhances the plate material flux into the bulk.
This paper analyses possible operational space for DTT device with liquid Sn or Li divertor setup. The impurities originating from the sputtering and vaporization processes are expected to modify plasma characteristics significantly both in the bulk and in the scrape-off layer. Therefore, simulations are performed with COREDIV code which self-consistently solves radial 1D energy and particle transport equations of plasma and impurities in the core region and 2D multifluid transport in the SOL.
Density and power scans are carried out for different target arrangements, in terms of the coolant temperature and thickness of W substrate. First scenarios with only intrinsic impurity are investigated. Tin appears more promising of lithium in terms of radiative capacity, of wider ranges of applicability both of density and input power and of plasma purity. No clear detachment is observed for either Sn or Li except at very high density. For both solutions regime where evaporation overcomes sputtering is more effective in dissipating the input power, provided that is kept low enough to ensure plasma stability. In this case a sort of vapour shielding seems to develop attached to the impurity source.

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14 MeV neutron streaming calculations for JET-like maze entrance using TRIPOLI-4 Monte Carlo code

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The International Thermonuclear Experimental Reactor (ITER) is currently under construction at Cadarache in southern France. The Joint European Torus (JET) is presently the largest tokamak in the world and the only one capable of using tritium. At JET, D-T fusion experiments will be conducted in 2019 (DTE2) on addressing the future ITER needs and reducing the risks of ITER operations. During the DTE2 operations, 14 MeV fusion neutrons will be generated. One of the aims of the DTE2 neutronics experiments is to investigate the D-T neutron streaming along the penetrations of biological shield of the JET Torus Hall. Continuous-energy Monte Carlo (MC) neutron transport calculations and TLDs measurements were already performed by different teams for the previous D-D neutron streaming benchmark experiments around the JET maze entrance. Using two-step simulation approach, two different calculation ways were performed to decrease the variances of calculated neutron fluence and ambient dose equivalent. Both the MC-MC approach using a midway surface source and the Deterministic-MC one using deterministic improved weight windows produced overestimated C/E results. The TRIPOLI-4 Monte-Carlo transport code has been extensively used on fission neutron radiation shielding analyses. To develop the TRIPOLI-4 application in ITER fusion neutronics, both experimental and computational benchmarks are being performed. In this study, using single-step approach combined with variance reduction (VR) techniques, TRIPOLI-4 shielding calculations were performed for a JET-like maze entrance in order to investigate the D-T neutron streaming from a Torus Hall of 40 m x 40 m x 4.2 m model. The calculation results including dose rate maps, the VR performance, and the user-friendly VR input of TRIPOLI-4 code will be reported in the final paper.

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Progress on in-vessel poloidal field coils optimization design for alternative divertor configuration studies on the EAST tokamak

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An upgrade to the lower divertor is currently being planned for EAST superconducting tokamak, aiming at >400s long-pulse H-mode operations with a full metal wall and a divertor heat load of ~10MW/m2. A new divertor concept for EAST, "Tightly Baffled Divertor", suited to water-cooled W/Cu plasma face components (PFCs) with minimized divertor volume, has been proposed to achieve Te,target<5eV across entire outer target at lower separatrix plasma density and optimized pumping by a simple closed divertor structure combining horizontal target with inclined baffle, dome and duct. This divertor should allow access to high-triangularity small-ELM H-mode regimes and also allow achieving Snowflake or X-Divertor like configurations with the assistance of two water-cooled in-vessel divertor coils (DCs). Preliminary engineering design of in-vessel DCs indicates a maximum current of 8kAt for long-pulse discharges, and 20kAt for the shortest ones. However, flexibility on DCs position optimization is limited to the water cooling system. Initial plasma equilibrium studies by FIXFREE code, used in combination with CREATE-NL and EFIT tools, show that the distance of the two nearby divertor poloidal field nulls, can be decreased up to ~0.8m with a plasma current IP~400kA, leading to a configuration with the secondary x-point located between the primary x-point and the target or close to the target, with a significant increase of magnetic poloidal flux expansion and connection length. This may provide a promising divertor solution compatible with advanced steady-state core scenarios.

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Heating and in-vessel upgrades of the TCV tokamak

Two sets of upgrades are being implemented on the TCV tokamak. The first set involves the installation of neutral beam injection (NBI) and new Electron Cyclotron (EC) power sources, to heat the ions and vary the electron to ion temperature ratio, for ITER relevant β values. A tangential 15-30keV, 1MW, 2s NBI is operational on TCV since 2015. A second 1MW, ~50keV beam, also tangential but opposite to the first beam, is foreseen to approach β limits, vary the momentum input and investigate suprathermal ion physics. On the EC side, two 0.75MW gyrotrons at the 2nd harmonic have been installed. The next step, presently ongoing, entails two 1MW dual-frequency gyrotrons (2nd and 3rd harmonics). The total heating power for high-density plasmas will increase from 1.25MW to 5.25MW. The main element of the second set is an in-vessel structure to form a divertor chamber, reach high neutral density and impurity compression and access reactor relevant regimes for a wide range of divertor configurations. Graphite gas baffles will be installed to separate the main chamber from the divertor. The first set of baffles consists of 32 tiles on the high-field side (HFS) and 64 tiles on the low-field side (LFS). The LFS tiles' dimensions will be varied to modify the divertor closure. Control of the plasma, neutral and impurity densities, and of the He compression in the divertor will be achieved by a combination of toroidally distributed gas injection valves, impurity seeding, and cryo-condensation pumps. Significant developments will be undertaken in plasma diagnostics, to characterize the divertor plasma, measure power and particle deposition at the strike points, and monitor the detachment process. The possibility of installing dedicated high temperature superconducting coils, to expand the range of divertor configurations and improve their control, will be discussed.

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Development of an electrochemical sensor for hydrogen detection in liquid lithium for IFMIF-DONES

Co-author: Nils Holstein [1]

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The structural materials in fusion reactors as DEMO and future power plants are under strong irradiation and will suffer from radiation damages. The knowledge of the radiation induced degradation is planned to be investigated in IFMIF-DONES, a facility in which fast neutrons are produced by a reaction of a D-beam with a liquid lithium target. The operation of such a system requires the control and measurement of impurity concentrations in the melt, thereunder hydrogen. Electrochemistry offers diagnostic tools to measure directly concentrations in such media by online-monitoring systems. Based on this technology, an electrochemical H-sensor for operation in liquid lithium is under development.

This presentation will line-out the physical background of measuring non-metallic impurity levels in molten metals by measuring electrochemical potentials and their transformation by Nernst correlation into concentrations. Liquid lithium is a very reactive melt, thus material section will be an essential topic for development of a reliably working sensor together with the materials, which can be used as hydrogen-conducting electrolytes in such a sensor. This material behaviour and properties will be discussed. Based on these issues a sensor design for hydrogen in liquid lithium was set up. The successful manufacturing and assembling of the sensor will be shown beyond the synthesis of the electrolytes which are essential for a pre-qualification of the sensor in liquid lithium. These tests are conducted under cleaned Ar atmosphere in a glove-box system as well as the whole sensor assembling. The outlook will deal with measurements in lithium of different hydrogen concentrations.

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Three-dimensional model of DEMO-FNS reactor for neutronics calculations and radiation shield problems

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At the present stage of the Demonstration Fusion Neutron Source (DEMO-FNS) design the actual problem is a development and use of the three-dimensional model of this device to the solution of various neutronics problems for the integration of the basic technological systems of tokamak. The radiation safety and the development of the radiation shield are crucial problems which significantly affect the layout and cost of the installation. Besides the basic radiation shield, the FNS blanket obtains shielding properties. The blanket design is currently being developed. The blanket is also a powerful source of neutrons. The recent neutronics studies of the blanket showed a crucial influence of a coolant substance on the rate of transmutations of the transuranium elements and the generation of tritium in the system. This paper presents the three-dimensional DEMO-FNS model developed for the Monte Carlo calculations and the results of its application for the estimation of the characteristics of the DEMO-FNS radiation shield to protect the materials of the superconducting coils of toroidal magnetic field (TMFC) from the neutrons and secondary photons, taking into account the replacement of the water coolant of the blanket to the carbon dioxide one. It was performed the analysis of the neutron balance, the neutron energy spectra and energy release of the neutrons and secondary photons in the shield, case and superconductor on the inner and outer contour of TMFC. It was found that the energy release in the case and superconductor on the outer contour of TMFC located between the injector port and the blanket maintenance port was overestimated in the area facing the injector port by about 10 times in comparison with the monolithic shield option. This indicates that the increase of the local iron-water shield of the injector port is required.

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Innovative and emerging melting technologies for fusion power plants wastes recycling

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To fully maintain the promise of energy production in clean, safe and environmentally responsible way, the nuclear fusion technology must include specific recycling and clearance techniques.
A number of options have been already proposed and investigated to recovery valuable elements, to separate radioactive species, to re-use materials in re-fabrication of components.
Melting based techniques, where materials after use are treated at high temperature and melted, are considered promising candidates, because the melt can be cast in a more compact volume for easier handling, disposal or for producing new components. The melt can be treated in appropriate conditions in order to remove undesired elements or to add species to recovery some specific characteristics, for further re-fabrication.
The most investigated melting options have been derived from well consolidated processes and plants widely applied in industrial sectors for mass production, such as steelmaking, glass, ceramic, foundry industries. The idea is to transfer these technologies taking advantage for the treatment and recycling of nuclear wastes. However the specific cases of radio-activated wastes expected from fusion power plant require significant and accurate re-design of technological solutions and operating conditions to make effective these options.
This paper presents and discusses examples of innovative technologies that could be fruitfully applied in melting processes for fusion power plant wastes treatment. The presented panorama includes technologies already applied at industrial level, in highly specialized, niche sectors, such as skull crucible technique, as well as laboratory and conceptual technology, such as magnetic levitation, still under development, but probably available when fusion power plant will be in service.
The aim of the paper is twofold: to enlarge the spectrum of melting technologies candidate for the needs of fusion power plant waste treatments: to help the individuation of necessary adaptation and improvement of these technologies for the specific case of radio-activated wastes.

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Delphi Exercise on the possible role of fusion energy in the global energy system

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The Delphi Exercise on the possible role of fusion energy technology in the future global energy system is a research project funded by the EUROfusion Socio-economic Studies (SES) programme. European research on fusion energy has a long tradition of energy scenario modelling as a way to study the economic and social conditions under which fusion would eventually become an energy technology option in the future. Previous research under the SES programme has emphasised for EUROfusion the importance of participatory socio-economic research with the inclusion of stakeholders from informed civil society. Following this, three reflection groups with civil society participants were organised:

- on the use of the concept of sustainable development in energy governance,
- on the use of modelling in energy foresight research and
- on the use of storylines in foresight research in general.

This Delphi Exercise should be seen as the fourth foresight research activity concerned with researching the possible role of fusion energy technology in the future global energy system. The presentation will discuss the results of the first round in which experts are invited to give their views on four points of attention:

1. aspects of future energy policy;
2. the use of reference scenarios such as those of the Intergovernmental Panel on Climate Change or the World Energy Council;
3. the use of storylines as a mean to construct quantifiable scenario's;
4. the use of drivers such as population growth, climate change, direct energy cost or technology availability.

In conclusion, the presentation will elaborate on how the results of this exercise can inspire future technological and social sciences research on fusion energy.

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Kinetics of double strand breaks of DNA in tritiated water evaluated using single molecule observation method

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Detailed understanding of mechanisms underlying DNA damages by low energy beta-rays from tritium is important for evaluation of impact of tritium release from fusion devices to the environment. In this study, the rate of double strand breaks (DSBs) of DNA in tritiated water was measured using a single molecule observation method.

Genome size linear double strand DNA molecules of bacteriophage T4 GT7 (57 micrometers, 166 kbp = kilo base pairs) were provided by Nippon Gene Ltd., Japan. The DNA molecules were suspended in sterilized and non-sterilized tritiated water (5.2 MBq/cm3) and irradiated with beta-rays at 10 ℃ for 1, 7 and 14 days. The values of absorption dose were 0.41, 2.9 and 5.7 Gy. After staining with fluorescent dye YoYo-1, DNA molecules were placed on a glass substrate in extended forms. The lengths of DNA molecules were measured using an inverted type fluorescent microscope (Olympus IX73). The number of DSBs were evaluated using a method proposed in [1]. DSBs in non-radioactive sterilized and non-sterilized water were also examined for comparison.

Under sterilized conditions, significant reduction in DNA lengths due to beta-ray irradiation was observed after irradiation for 7 and 14 days corresponding to 2.9 and 5.7 Gy, respectively. The number of DSBs per 100 kbp was evaluated to be 0.2 at 1 day, 1.0 at 7 days and 1.6 at 14 days. Interestingly, DNA lengths in non-sterilized tritiated and non-radioactive water were clearly shorter than those in sterilized tritiated water. It means that the irradiation effects of beta-rays were far weaker than the influence of microorganisms in water though tritium concentration was as high as 5.2 MBq/cm3.

[1] S. F. Shimobayashi et al., J. Chem. Phys. 138, 174907 (2013).

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Analysis of technical and economic parameters of fusion power plants in future power systems

Co-author: Inga Maria Müller [1]

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To restrict climate change, it is highly desirable to replace conventional power plants by emission free technologies like solar and wind power. This leads to increased shares of fluctuating sources in the power system challenging the balance between generation and demand. Moreover, as the transportation and heating sectors are expected to shift towards using more electricity the projected demand rises strongly. To ensure both sufficient supply and long-term stability of power systems, fusion power may play an important role in the future due to its advantageous properties. Fusion energy is emission free, controllable and inherently safe.

In recent years a lot of research has been conducted focusing on the technical principles of commercial fusion power. However, less investigation has been done regarding the interaction of different future power plants with the power system.

This paper provides a detailed analysis of technical and economic parameters of fusion power plants which are relevant in context of power system such as efficiency, heat losses, investment costs and startup costs. Therefore, two linear optimization models – urbs and evrys – are applied which both represent the European power system by including power plants, storages, transmission lines and loads of 34 countries. Both models optimize the total system costs. urbs is applied for expansion planning whereas evrys present a detailed dispatch in hourly resolution. As a result, clear statements about system costs, system configuration and system operation are made based on a comprehensive scenario framework.

First results indicate, that investment costs and fixed costs of fusion power plants as well as emission restriction highly affect investment decisions. The more flexible a fusion plant can operate, the more fusion power will be installed. The operation of fusion power plants is more dependent on efficiency rates and startup costs than on thermal properties like cooling and heating.

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Uncertainty analysis of an SST-2 fusion reactor design

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Systems codes are a powerful tool for designing the next generation of nuclear fusion reactors. By exploring a large design space in a single calculation, they can obtain highly optimised solutions. However, while a single design is informative, it does not give the whole picture. Often new designs will push boundaries, whether that involves scaling to new physical regimes or applying new technologies. All of this will introduce uncertainty which needs to be quantified to give a complete understanding of the performance of a proposed reactor. Uncertainty analysis can then inform about high impact areas, critical design aspects or simply confirm the robustness of the design.

The SST-2 reactor is a proposed medium sized device with low fusion gain (Q = 5) and capable of producing fusion power from 100 to 300 MW. Tritium breeding will be achieved by having breeding blankets only on the outboard side, while on the inboard side, shielding blankets will be placed due to the limited space available. The magnets will be superconducting in nature to achieve steady state operation. In this work, we are performing an uncertainty analysis of its latest published concept design (Srinivasan et al. 2016; Fusion Eng. Des., 112, 240). We initially use the systems code PROCESS to reproduce the SST-2 design originally obtained using another systems code, SPECTRE, highlighting and explaining the differences between the two. We then apply a Monte-Carlo based uncertainty quantification tool using PROCESS to explore its predicted performance and highlight high impact areas. By building a more comprehensive understanding of the implications of the uncertain parameters we can give better predictions to aid the development of the design.

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ANITA-NC: a Code System for Modelling Material Activation Induced by Neutral or Charged Particles

Co-author: Manuela Frisoni [1]

[1] *FSN ENEA*

The evaluation of the neutron induced material activation plays an important role for the development of future fusion power plants for issues related to safety, engineering design and radioactive waste management. For these devices the activation codes and cross section libraries handling neutron energies up to 20 MeV are quite adequate.
Besides, in order to study the irradiation effects on fusion materials, some facilities have been proposed to produce accelerator-based neutron sources of sufficient intensity to test samples of candidate materials to be used in future fusion power plants. The International Fusion Materials Irradiation Facility (IFMIF) and, more recently, DONES (DEMO Oriented Neutron Source), more tightly focused on DEMO needs, have been proposed to be such dedicated facilities. In both these plants the neutron source is produced through the reaction of 40 MeV deuterons impinging on a liquid lithium target so that neutrons with energies up to 55 MeV are produced.
In these facilities the deuterons will themselves cause activation, particularly in the accelerator structure and in the lithium target.
ANITA-NC (Analysis of Neutron Induced Transmutation and Activation – Neutral and Charged) is an inventory code developed in ENEA-Bologna, capable of modelling material activation induced by several neutral and charged projectiles (neutron, proton, alpha, deuteron and gamma). It is an extension of the previous ANITA versions.
ANITA-NC uses the EAF-2010 group-wise cross-section libraries in EAF format. The VITAMIN-J+ (211) energy group structure for neutrons, up to 55 MeV, and the CCFE (162) group structure for p, α, d and γ, up to 1 GeV, are used.
The decay data used in ANITA-NC are based on the JEFF-3.1.1 Radioactive Decay Data Library.
The present paper summarizes the main characteristics of the ANITA-NC activation system. Some relevant application cases and examples of data and code validation on experimental results are also shown.

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In-box LOCA accident analysis for the European DEMO water-cooled reactor

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Transient analysis in a water-cooled fusion DEMO reactor have been performed to support the WCLL (Water-Cooled Lithium Lead) breeding blanket design. In this framework, the Design Basis Accident analysis of an in-box LOCA has been carried out.
The WCLL breeding blanket concept relies on Lithium Lead (LiPb) as breeder, neutron multiplier and tritium carrier, which is cooled by water at 15.5 MPa with an inlet temperature of 295℃ and an outlet temperature of 328℃.
Water flows in Double-Wall Tubes (DWTs) in order to reduce the probability of water/LiPb chemical interaction. In the case of a LOCA accident, multiple rupture of these tubes is postulated, with consequent leakage of pressurized water in the LiPb side of the module.
The present safety analysis has been performed with the MELCOR computer code (ver. 1.8.6) modified for the application in the fusion context. Custom models are employed to simulate the chemical water/LiPb interaction in the module. The rupture mass flow rate calculated in water simulation is transformed in its equivalent in terms of hydrogen and unreacted water steam. Both have been treated as non-condensable gas. Two different input decks, one for each fluid considered, have been coupled through an external interface to account for their reciprocal interaction.
Pressure and temperature transient behavior in the broken module demonstrate that safety margins are respected during the entire accidental sequence, even though no external safety system is foreseen or actuated. Moreover, particular attention has paid to the quantity of hydrogen produced in

order to support the development of solutions, suitable for the DEMO, to prevent hydrogen explosion.

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The added value working under ISO 9001 in nuclear fusion technology R&D at ENEA

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The ENEA Fusion Department (FSN) operates in the field of nuclear fusion under a Quality Management System (QMS) according to ISO 9001 since 2011. At the beginning this methodology applied in R&D activities of a Research Institution such as ENEA seemed to be far from the industrial reality according to an internal and external perspective. But now that the construction of ITER reactor became a reality for which the industry and the R&D activities are overlapping, the ISO 9001 approach is becoming a winning solution. Many ITER tenders require that applicants have a certified QMS. In addition to the higher potential to achieve success in these tenders, having a QMS produced in ENEA FSN positive results in terms of added value in planning, risk-based thinking, control services, document management, metrology, performance evaluation and continual improvement. These actions defined by appropriate procedures are useful to satisfy the customer, but also to improve the internal organization of work. The methods applied to obtain the sensitivity to the critical issues and to spread the culture of quality have been adapted to the unique sector of nuclear fusion technology (e.g. design and manufacture of specific components as the divertor HHF components or superconducting magnets, performance of experiments, etc.). The references for these methods are still the performance assessment through the monitoring, measurement, analysis and evaluation of processes and internal audits and the pursuing of continual improvement, but the indicators used for processes measurement are not those typical of a company in which most of the indicators are linked to the economic profit, but some more specific; the impact factor of research activities, the self-assessment efficiency, the continual training of the personnel, the management of resources for measurement, the assessment of the value of ENEA FSN activities by the analysis public press.

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Material optimization technique to minimize radiological responses in fusion reactors

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High-energy, high-intensity neutrons emitted from the fusion plasma present a stringent environment for the structural materials present in the fusion device. This has significant life-limiting effects on the reactor components. The neutrons interact with the material initiating nuclear reaction leading to the production of radioactive isotopes, gas molecules and material defects. These gases, particularly helium, can cause swelling and embrittlement of the material. Furthermore, the radioactive isotopes produce would cause heating in the material. Some of these isotopes may have long lives which would contribute towards the radwaste produced in the fusion devices. Hence designing of low activation materials for fusion devices is warranted.
At Iter-India, Institute for Plasma Research a number of computational tools are being developed to estimate the nuclear response of the materials. ACTYS-1-GO, a multipoint neutron activation code which can calculate radiological responses of materials located at various positions in a fusion reactor efficiently, is developed. Along with it, a mathematical framework is developed for accessing the relationship of radiological quantity with the initial elements present in the material. Such framework helps in identifying and thus minimizing the elements/isotopes from the initial material

composition.
In the present study both the methodologies are efficiently coupled for a complete material optimization. Quantities responsible for various radiological effects (like activity, dose, heat, and radwaste) and related defects in the material are at all irradiation times considered and their contributing elements are sorted and minimized accordingly. Also, since a single material faces a gradient of neutron flux over its entire volume, all such optimization is carried out over the entire range of neutron flux faced by that material. This would provide the best material composition in accordance with the neutron environment faced by the material and its functionality, minimizing the radiological effects like activity, dose, gas production and radwaste.

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Laser Technology for Resident Tritium In-situ Measurement in CEPT Device

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Due to the fact that during Tokamak operation（Plasma Facing Materials（PFM）are able to trap part of the fuel（particularly Tritium）, these resident fuel have to be measured and removed. LIBS（Laser induced breakdown spectroscopy） and LIDS （Laser induced desorption spectrometry）are two of the most promising techniques to solve these issues which allow to achieve an on-line and ultra-sensitive measurement of the fuel retention in vessel.
In this work, LIBS and LIDS were integrated on to a Tritium penetrating and resident assessing instrument named CEPT（Comprehensive ECR Plasma for Tritium） for in-situ tritium retention analysis. An Nd:YAG laser, operating at 1064 nm, with pulse duration of 4-7 ns was used to interact with PFM simulant sample. The spectra emitted from plasma plumes was recorded by the Aryelle 200 spectrometer(/=12500) in the spectral range between 265 nm and 665 nm with an Andor iStar 334 ICCD camera. The mass spectra of particles escaped from the sample due to the laser radiation were obtained using a quadrupole mass analyser (QMA) at range of 1-100 amu.
A verification experiment was carried out at a pressure range from 1 bar to ultra-high vacuum(10-5 Pa) in He and air for LIBS and range from 10-2-10-5 Pa for LIDS. The LIBS results show that the atomic spectral lines of H, D, W was achieved when a delay time of 0.3 us was installed, in the meanwhile, the released hydrogen isotopes were monitored by the QMA. This technique achieves good spatial resolution without any sample preparation. For comparison, the samples were also analysed by other conventional methods.

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P4.144 Thermal-hydraulic analysis for first wall and vacuum vessel thermal shield of Divertor Tokamak Test facility

The Divertor Tokamak Test (DTT) machine has been proposed by ENEA, in collaboration with other Italian institutions, to investigate power exhaust solutions with an experiment integrating all DEMO relevant physics and technology issues. The DTT machine will be able to host, in different phases of its life-time, advanced divertor magnetic configurations (snowflake, super-X, double null) and liquid metal solutions, able to withstand the large loads expected in the DEMO fusion power plant. The first wall (FW) has been designed with stainless steel cooling pipes coated with a W layer deposited by plasma spray technique. It shall be compatible with liquid metal divertors and therefore be heated at a temperature of 300 ℃ to avoid the liquid metal condensation. In this case it is necessary to consider gas rather than water cooling. To minimize the heat transferred to the superconductive coils the vacuum vessel (VV) is kept at 100 ℃ with water and a Vacuum Vessel Thermal Shield (VVTS) cooled with helium at 70K is interposed between the VV and the coils. In this work the preliminary results

of the thermal-hydraulic analysis carried out with ANSYS Workbench software are presented for both the FW, using CO2 as coolant, and the VVTS.

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Plasma control for EAST long pulse non-inductive H-mode operation in a quasi-snowflake shape

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Advanced magnetic divertor configuration is one of the attractive methods to spread the heat fluxes over divertor targets in tokamak because of enhanced scrape-off layer transport and an increased plasma wetted area on divertor target. Exact snowflake (SF) for EAST is only possible at very low plasma current due to poloidal coil system limitation. However, we found an alternative way to operate EAST in a so called quasi-snowflake (QSF) or X-divertor configuration, characterized by two first-order nulls with primary null inside and secondary null outside the vacuum vessel. Both modeling and experiment showed this QSF can result in significant heat load reduction to divertor target [1]. In order to explore the plasma operation margin and effective heat load reduction under various plasma conditions and QSF shape parameters, we developed ISOFLUX/PEFIT shape feedback control. In experiment, we firstly applied the control of QSF in a similar way to control the single null divertor configuration, with specially designed control gains. Reproducible QSF discharges have been obtained with stable and accurate plasma boundary control. Under Li wall conditioned, we have achieved highly reproducible non-inductive steady-state ELM-free H-mode QSF discharges with the pulse length up to 20s, about 450 times the energy confinement time by using low hybrid wave, ion cyclotron resonance wave (ICRH) and electron cyclotron resonance wave (ECRH) for the plasma current drive and heating. The capability of the QSF to reduce the heat loads on the divertor targets has been confirmed. This new steady-state ELM-free H-mode QSF regime may open a new way for the heat load disposal for fusion development.

O1.C / 12

Deep Learning: Towards Autonomous Remote Maintenance

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Traditionally, remote maintenance in fusion and other nuclear plants has made use of man-in-the-loop telemanipulator devices in order to deal with the relatively unpredictable nature of tasks, and complex environments. Future fusion devices will require maintenance orders of magnitude more complex than at present, however it is infeasible to scale remote maintenance operations teams linearly with the increase in device complexity.

Combined with increasing demand for productivity it will therefore be necessary to automate large numbers of maintenance tasks, many of which have previously been reliant on human-level dexterity and intelligence. This has previously been infeasible due to limitations of automation technology, however recent developments in artificial intelligence are showing a great deal of promise.

The new and rapidly advancing field of deep learning has developed a number of advanced machine learning techniques which have not only surpassed the performance of previous methods, but also, in some cases, outperform human-level performance in a number of challenging task areas. We present recent developments in deep learning which are relevant to nuclear fusion, as well as a range of research activities which have been taking place at RACE related to deep learning for automation of robotic tasks in fusion environments. We describe how these new techniques are changing what can be considered possible in remote maintenance and how methods for remote maintenance are evolving.

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Myth of Initial Loading Tritium-2 : Practical commissioning strategy of DEMO-Japan without external source

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The authors have pointed out that initial tritium needed for starting operation of fusion reactor can be made by DD and low T discharges with self sufficient blankets. Practical commissioning plan of Japanese DEMO was recently planned as a part of DEMO design activity. The early campaigns will require longer than a year of repeated low power pulses for operational purposes as the "power ascension tests". Breeding blankets with TBR well above unity is designed based on a water cooled ceramic pebble concept. Complete tritium plants should be continuously operated with torus exhaust and blanket recovery for safety reasons. In the phase 0 of commissioning, DD pulses with small flux neutron followed by relatively long dwell periods are used to confirm all the nuclear functions of DEMO plant including the Balance of Plant and ancillary systems. This study analyzed dynamic tritium behavior in the plant. After the each discharges, produced tritium from DD reaction in the plasma, bred in the blanket, and fed to the vacuum vessel are all collected by the tritium plant and recovered from the isotope separation during the discharge during the dwell periods. After the DD phase, small amount of tritium is added to the fuel, however the required amount if gram level, that is available from the storage in the tritium plant. In the later operation, tritium concentration will gradually increase and the pulse length will longer, however for each shots, sufficient tritium can be prepared prior to the burning. It was found that in this commissioning scenario, no external tritium or additional DD shots for tritium production is required for DEMO program in Japan.

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Implementation and exploitation of jet enhancements in preparation for dt operation and next step devices

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JET presents some unique capabilities: the reactor fuel, ITER wall materials and the capability to confine the alphas. JET next T-T and D-T experimental campaigns can therefore address major physics and technological gaps for the development of fusion energy: the isotopic effects on confinement, the access the H mode and ELM behaviour. The total yield of the final D-T phase is expected to be 1013 n/s·cm2, a factor of six higher than the previous DTE1. In this context, three main aspects of JET capability have been recently improved: 1) scenario development to enhance performance 2) the quality of the measurements to maximize the scientific exploitation 3) specific technologies for ITER and DEMO.
With regard to the scenarios, the performances of JET with a carbon wall have been reproduced up to a current of 3 MA, which supports the prediction of 15 MW fusion power in full DT. Moreover

improved control systems (wall load protection, simultaneous control of ELM frequency and beta, plasma mixture) insure that the plasma configurations are compatible with the wall properties, from melting to retention and dust production.
In terms of general diagnostic capability, JET can now deploy much better resolution diagnostics, particularly for the edge quantities, and a consistent set of techniques to diagnose the fast particles, from redistribution to losses, using techniques ranging from gamma ray spectroscopy to a scintillator probe and Faraday cups. A full calibration of the neutron diagnostics for the 14 MeV neutrons has just been completed successfully.
With regard to ITER and DEMO relevant technologies, specific programmes are being pursued to investigate: the radiation field, the induced activity and dose rates and the radiation damage of materials. Dedicated studies for DEMO, including the tests of a new tritium pumping cycle and a tritium breeding blanket mock-up, are also almost completed.

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Manufacturing of the first ITER Torus Cryopump

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The first of nine ITER Torus and Cryostat Cryopumps has been successfully manufactured and delivered to ITER in summer 2017. This Pre- Production Cryopump is the first of the ITER cryopumps and may be used for the first pump down of the vacuum vessel or the cryostat.
The pump has a 1.8 m diameter and a length of about 3 m and contains cryogenic pressure equipment with a charcoal coated adsorption stage integrated in a casing with a vacuum vessel plug combined with the largest all-metal vacuum valve ever built resulting in an overall weight of 8 tons. The design of the cryopump is the result of more than ten years of research and development finalized to the ITER built to print design to comply with the demanding requirements to be fulfilled during its operation starting from first plasma.
The paper will outline the experience gained with the cryopumps manufacture and assembly. We discuss the components with confinement function as ITER style vacuum flanges, double bellows for the valve assembly and electrical feedthroughs. Many different requirements had to be addressed for the manufacture of these components and their integration in the cryopump.
ITER is a Nuclear Facility, INB-174, and requirements for the operation in the primary vacuum and the nuclear confinement function demand a high level of quality control and inspection needs during all manufacturing stages. The Pre-Production Cryopump has been built in close and successful cooperation with F4E reflected in the adequate surveillance of the 27 suppliers required to fabricate the cryopump. The successfully built and delivered Pre-Production Cryopump will give a reliable basis for the F4E procurement of the eight Torus and Cryostat Cryopumps which are required for first plasma.

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

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Reconstructing JET using LiDAR-vision fusion

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The containment vessel of the Joint European Torus is a huge, complicated assembly with a myriad of components, all of which are important for plasma operation. As a research device, JET has been operated over many years and has been extensively rebuilt. During each maintenance shutdown, inspections and measurements of the Vacuum Vessel are carried out by means of dual-camera Stereo surveys, High-Resolution single camera surveys and precise Gap Gun measurements. This is a precise but labour-intensive process, taking tens of hours to complete a full survey.

Due to rapid advancements in the field, combined visual-LIDAR techniques have evolved to the point where it is possible to carry out on-line, high-resolution measurements of the interior of buildings and scientific installations. Since the radioactivity inside the JET vessel is still low enough to allow consumer-grade electronics to survive unprotected, these advancements can be leveraged.

We present work including the 3D mapping of the inside of the JET Torus using a combined LIDAR-Vision measurement and navigation system. Using one of the remote handling booms, we carry out a scan of the JET vessel. We compare the point cloud model with the CAD models of the JET installation using numerical methods, demonstrating mm and sub-mm accuracy with a dramatically lower survey duration compared to existing techniques. We also compare the estimated path of the scanner through the vessel with the recorded boom joint position data. Conclusions are drawn about the applicability of LIDAR systems to mapping and localisation problems within a Fusion environment as well as assessing the resulting accuracy of the scan.

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Engineering and integration design risks arising from advanced magnetic divertor configurations

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The divertor configuration defines the power exhaust capabilities of DEMO as one of the major key design parameters and sets a number of requirements on the tokamak layout, including port sizes, PF coil positions, and size of TF coils. It also requires a corresponding configuration of plasma-facing components and a remote handling scheme to be able to handle the cassettes and associated in-vessel components the configuration requires.

There is a risk that the baseline ITER-like single-null (SN) divertor configuration cannot meet the PFC technology limits regarding power exhaust and FW protection while achieving the target plasma performance requirements of DEMO or a future fusion power plant. Alternative magnetic configurations - double-null, snowflake, X-, and super-X - exist and potentially offer solutions to these risks and a route to achievable power handling in DEMO. But these options impose significant changes on machine architecture, increase the machine complexity and affect remote handling and plasma physics and so an integrated approach must be taken to assessing the feasibility of these options.

In this paper we describe the work being undertaken, and main results so far, in assessing the impact of incorporating these alternative configurations into DEMO whilst respecting requirements on remote handling access, forces on coils, plasma control and performance, etc.

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Electromagnetic FEM studies of disruptions and engineering consequences for the power supply and coils design of planned upper divertor at ASDEX Upgrade

Co-authors: Markus Teschke [1]; Albrecht Herrmann [1]; Gabriella Pautasso [1]; Thomas Vierle [1]; Irene Zammuto [1]

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There is proposed a new upper divertor for the ASDEX Upgrade tokamak experiment [1]. It is planned to be equipped with internal coils for investigation of advanced magnetic configurations like e.g. „snowflake“. Due to the close vicinity of the coils to the plasma, high induced and very stiff voltages are expected during disruption events. Because only very vague analytical estimates of voltages, forces and coupling factors were available, an improvement by the help of finite element method (FEM) was envisaged. Therefore, recorded measurements of currents, plasma position, plasma profile and the geometry were integrated into the electromagnetic simulation as boundary conditions to calculate resulting field distributions during selected AUG disruption events. The time resolution can be better than 100 microseconds and the required computing resources are comparable small due to utilization of 2D axis-symmetry. The results were compared with magnetic probe measurements integrated into the tokamak. They are in good agreement. After this, the simulated geometry was modified to the target geometry including the new divertor to calculate all relevant parameters. The output of these calculations has strong implications for the coil and power supply design: (1) The power supply will be protected with a new kind of crowbar to avoid uncontrolled current and force rise of the coils and power supply damage due to overvoltage. The concept of this so called “ripping crowbar” will be introduced, which is under development, now. (2) The coil cable should be coaxial shaped to monitor isolation faults and to become inherently safe against single-turn shortcuts, identified as a destructive fault scenario.

[1] A. Herrmann, et al., Fusion Engineering and Design 123 (2017) 508-512.

O1.B / 50

Neutronics of the IFMIF-DONES irradiation facility

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Within the Early Neutron Source (ENS) project of EUROfusion the design of the accelerator based irradiation facility IFMIF-DONES (International Fusion Material Irradiation Facility- DEMO Oriented NEutron Source) is under development. The main mission of IFMIF- DONES is to provide the irradiation data needed for the construction of DEMO, a fusion power demonstration reactor developed in the frame of the Power Plant Physics and Technology (PPPT) programme of EUROfusion.

The IFMIF-DONES facility consists of a deuteron accelerator, a liquid lithium target and a Test Cell with irradiation test modules as main systems. Neutronics has to provide the data which are required

to design and optimize these systems, evaluate and prove their nuclear performance, and ensure a sufficient radiation protection. In addition, the radioactive inventories, produced during operation, have to be assessed to enable sensible maintenance and waste management strategies.

A variety of neutronics simulations is needed to compute the nuclear responses for all systems and components and provide the radiation fields during operation, maintenance and shut-down periods. Such simulations require dedicated computational approaches adapted to the needs and peculiarities of the accelerator based IFMF-DONES neutron source. The ENS project thus builds on the development of specific tools and data for simulating the interactions of deuterons with the lithium target and the accelerator structures, the generation and transport of neutrons and photons, and the production of radio-active nuclides with the subsequent emission and transport of decay gamma radiation.

The paper presents an overview of the IFMIF-DONES neutronics comprising both nuclear analyses and the applied computational approaches. Main issues are the nuclear analyses conducted lately for the Accelerator Facility and the Test Cell utilizing the specific codes and data developed and/or adapted for IFMIF-DONES. Related R&D issues are also addressed.

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Methodology for reverse engineering analysis of ITER as-built integrated systems

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The ITER machine consists of a large number of highly integrated and complex systems, with critical functional positional requirements (e.g. accurate positioning of magnets to minimize error fields and location of plasma facing components with respect to magnetic axis) and reduced design clearances to maximize Tokamak performances and limit costs. Deviations from specified part tolerances and product assembly processes/accuracies could have a critical impact on assembly feasibility and achievement of final performances, compromising plasma operation. The assessment of potential deviations on as-manufactured and as-assembled systems, together with early mitigation of non-compliances with Tokamak dimensional requirements, are critical activities during the ongoing manufacturing and Tokamak construction phases.
This paper describes the detailed methodology implemented within ITER for the assessment of as-built integrated systems, based on Reverse Engineering (RE) techniques. The procedure includes the identification of construction needs (at system and assembly levels), the acquisition of metrology data based on defined needs, the post-processing and alignment of data and the reconstruction and management of 3D as-built configuration models. A detailed description of the assessment of non-conformities and tolerance risks/opportunities based on this methodology is included. Predictive clash detection and assembly process assessment/optimization are also described. RE techniques are a basic tool for the management of dimensional risk and opportunity for mitigation or recovery during ITER Construction phase.
ITER is a Nuclear Facility INB-174. This paper describes a Protection Important Activity (PIA) for safety

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Integrated current profile, normalized beta and NTM control in DIII-D

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There is an increasing need for integrating individual plasma-control algorithms with the ultimate goal of simultaneously regulating more than one plasma property. Some of these integrated-control solutions should have the capability of arbitrating the authority of the individual plasma-control algorithms over the available actuators within the tokamak. Such decision-making process must run in real time since its outcome depends on the plasma state. Therefore, control architectures including supervisory and/or exception-handling algorithms will play an essential role in future fusion reactors like ITER. However, most plasma-control experiments in present devices have focused so far on demonstrating control solutions for isolated objectives. In this work, initial experimental results are reported for simultaneous current-profile control, normalized-beta control, and NTM suppression in DIII-D. Neutral beam injection (NBI), electron-cyclotron (EC) heating & current drive (H&CD), and plasma current modulation are the actuation methods. The NBI power and plasma current are always modulated by the Profile Control category within the DIII-D Plasma Control System (PCS) in order to control both the current profile and the normalized beta. Electron-cyclotron H&CD is utilized by either the Profile Control or the Gyrotron categories within the DIII-D PCS as dictated by the Off-Normal and Fault Response (ONFR) system, which monitors the occurrence of a Neoclassical Tearing Mode (NTM) and regulates the authority over the gyrotrons. The total EC power and poloidal mirror angles are the gyrotron-related actuation variables. When no NTM suppression is required, the gyrotrons are used by the Profile Control category, but when NTM suppression is required, the ONFR transfers the authority over the gyrotrons to the NTM stabilization algorithm located in the Gyrotron category. Initial experimental results show the potential of the ONFR system to successfully integrate competing control algorithms.

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Innovative Technology for 6Li Enrichment using Electrodialysis with Lithium Ionic Conductor

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Tritium needed as a fuel for fusion reactors is produced via neutron capture by lithium-6 (6Li). However, natural Li contains only about 7.8% 6Li, and enrichment of 6Li up to 90% is required for adequate tritium breeding in fusion reactors. In Japan, lithium isotope enrichment methods have been developed to avoid the environmental hazards of using mercury. However, the isotope separation coefficient and efficiency is too low to meet the practical need of large mass production of 6Li.
Therefore, new Li isotope separation technique using a Li ionic superconductor functioning as a Li isotope separation membrane (LISM) have been developed. First of all, I investigated the ionic mobility of lithium isotopes in ionic superconductor. Combing the first principle and the kinetics Monte Calro simulation, I calculate the diffusion constant of 6Li and 7Li.
Furthermore, examinations of Li isotope separation using LISM with electrodialysis ware performed. Because the mobility of 6Li ions is higher than that of 7Li ions, 6Li can be enriched on the cathode side of a cell. Using Li0.29La0.57TiO3 (LLTO) as the Li ionic superconductor was prepared. After electrodialysis, I obtained a maximum of 1.05 for the 6Li isotope separation coefficient. This result showed that the 6Li isotope separation coefficient of this method is the same as that of the amalgamation process using mercury (1.06).

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Digital valve system for ITER remote handling – design and prototype testing

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ITER-RH system is used to exchange the divertor's 54 cassette assemblies in the vessel. Water hydraulics and servo valves are currently used in the task requiring high accuracy tracking and the use of de-mineralized water. The main concern has been robustness of the technology. Only few suitable commercial water servo valves exist and problems e.g. with jamming and wear been encountered. A possible mitigation is to use redundant valves but ensuring required level of water cleanliness is still an issue.

An alternative option is to use digital technology where on/off valves and intelligent control is used to produce proportional output. So far, no proper high-pressure on/off valves existed in the market and therefore a new concept was developed and tested with promising results. The valve has very fast response time, flow capacity that suits well for the required velocities and is compatible with de-mineralized water. In addition, the valve is not sensitive against water cleanliness and can be made rad-hard when necessary.

A mock-up of the remote handling system was used as test bench. The system was simulated in order to dimension the valve system and to tune the valve controller. A complete valve package with 16 prototype on/off valves and a manifold was manufactured and assembled. The rest of the components were off-the-shelf and the result is fully compatible to be used as direct replacement for the servo valve system.

Long-term tests of 60 working days and 2000+ hours was made. A new level of performance was demonstrated throughout the tests as tracking accuracies were approximately ten times better than with servo valve. Although some design faults where detected and system had faulty components, tracking accuracy was very good. Tracking and positioning capability of digital valve system exceeds all requirements and seems applicable for ITER-RH application.

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Conceptual design of the COMPASS-U tokamak

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The Institute of Plasma Physics of the CAS in Prague has recently started construction of new COMPASS-U tokamak. It will be a compact, medium-size (R = 0,85 m, a = 0,3 m), high-magnetic-field (5 T) device. COMPASS-U will be equipped by a flexible set of poloidal field coils and capable to operate with plasma current up to 2 MA and, therefore, high plasma density (~ 10^20 m^-3). The device is designed to generate and test various DEMO relevant magnetic configurations, such as conventional single null, double null, single and double snow-flake. The plasma will be heated using 4 MW Neutral Beam Injection (NBI) heating system with future extension by at least 4 MW Electron Cyclotron Resonant Heating (ECRH) system.
COMPASS-U will be equipped with lower and upper closed, high neutral density divertors. Due to high PB/R ratio COMPASS-U will represent a device which will be able to perform ITER and DEMO relevant studies in important areas, such as the plasma exhaust or development of new confinement regimes. The divertors will use conventional materials in the first stage, however, in the later stage, the liquid metal technology, which represents a promising solution for the power exhaust in DEMO,

will be installed into the lower COMPASS-U divertor. The metallic first wall will be operated at high temperature (approx. 300 ℃) during plasma discharge, which will enable to explore the edge plasma regimes relevant to ITER and DEMO operation. The first plasma is scheduled for 2022.
In this contribution, we will present the conceptual design of the COMPASS-U tokamak as well as the main tokamak components.

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Status of the EU DEMO breeding blanket manufacturing R&D activities

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The realization of a DEMOnstration Fusion Power Reactor (DEMO) to follow ITER, with the capability of generating several hundred MW of net electricity and operating with a closed fuel-cycle by 2050, is viewed by Europe as the remaining crucial step towards the exploitation of fusion power. The EUROfusion Consortium, in the frame of the European Horizon 2020 Program, is assessing four different breeding blanket concepts in view of selecting the reference one for DEMO. This paper describes technologies and manufacturing scenarios developed and envisaged for the four blanket concepts, including nuclear "conventional" assembly processes as TIG, electron beam and laser welding, Hot Isostatic Pressing (HIP), and also more advanced (from the nuclear standpoint) technologies as additive manufacturing techniques.
With regard to welding processes, topics as the metallurgical weldability of EUROFER steel and the associated risks or the development of appropriate filler wire are discussed.
The development of protective W-coating layers on First Wall, with Functionally Graded (FG) interlayer as compliance layer between W and EUROFER substrate, realized by Vacuum Plasma Spraying method, is also propounded. First layer systems showed promising layer adhesion, thermal fatigue and thermal shock properties. He-cooled mock-ups, representative of the First Wall with FG W/EUROFER coating will be fabricated for test campaigns in the HELOKA facility under relevant heat fluxes.
First elements of Double Walled Tubes (DWT) manufacturing and tube/plate assembly for the water cooled concept are given, comprising test campaign aiming at assessing their behaviour under corrosion.
Developments described in the paper are performed in conformity with international standards and/or design/manufacturing codes.
Eventually, further development strategies are suggested.

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Picosecond laser-induced ablation for depth-resolved analysis of first wall components in fusion devices

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Monitoring the fuel content of plasma-facing components is a key challenge for fusion devices like Wendelstein 7-X (W7-X) [1], equipped with graphite PFCs or ITER with beryllium/tungsten components. In the case of ITER, it is essential to limit the tritium content in the first wall to comply with safety regulations and to sustain the tritium cycle. In W7-X the measurement and control of the hydrogen content in the first wall mandatory to achieve stable long pulse operation. Laser-Induced Breakdown Spectroscopy (LIBS) and Laser-Induced-Desorption (LID) [2] are suitable to determine the fuel retention ex-situ in extracted PFCs or in-situ in the vacuum vessel during or between plasma discharges. However, the self-consistent quantification is challenging as the emission is a non-equilibrium process and often a comparative measurement with pre-characterized reference samples is mandatory.

A combination of LIBS and residual gas analysis is an alternative approach: For volatile sample components, the linearity of the quadrupole signal to gas pressure simplifies the calibration and reduces uncertainties. We use a Nd:YVO4-laser (λ3rd=355 nm) with a pulse duration of τ=35ps and pulse energies up to E=40mJ. With a spot size diameter on the sample of x=0.7mm we achieve a depth resolution of Δh=100nm for graphite samples without significant matrix mixing effects.

We present a series of composition analysis of poloidal and toroidal locations on graphite limiter tiles of W7-X, exposed to hydrogen plasma in OP1.1. Erosion- and deposition-dominated zones could be identified. In addition, graphite divertor tiles exposed in OP1.2a are analyzed. In contrast to ms-LID, the heat penetration depth of ps-laser-induced-ablation is smaller than the ablation rate. Consequently, all retained hydrogen from the sample is removed and quantitative depth-resolved hydrogen retention information (O(1022/m2)) is gained.

[1] T.S. Pedersen et al., Nucl. Fusion 55 (2015)126001.
[2] G. De Temmerman et al., Nuclear Materials and Energy (2016)2352-1791.

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Development of HINEG and its experimental campaigns

Co-authors: Yican Wu [1]; Yongfeng Wang [1]; Chao Liu [1]; Zhigang Wang [1]; Jieqiong Jiang [2]; Pengcheng Long [1]; Taosheng Li [2]; Liqin Hu [1]; Jianye Wang [1]

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Fusion energy becomes essential to solve the problem of increasing energy demands. A high intensity D-T fusion neutron generator is keenly needed for the research and development (R&D) of fusion technology, especially for fusion materials research.
The Institute of Nuclear Energy Safety Technology (INEST), Chinese Academy of Sciences (CAS) has launched the High Intensity D-T Fusion Neutron Generator (HINEG) project. The R&D of HINEG includes three phases: HINEG-I has been constructed and successfully produced a D-T fusion neutron yield of up to 6.4E12 n/s. The mechanism research of irradiation damage for materials can be carried out. HINEG-II aims at a high neutron yield of 1E15~1E16 n/s neutrons via high speed rotating tritium target system and high intensity ion source, which could be used to conduct material irradiation damage testing. The preliminary design and research on key technologies are on-going. HINEG-III is a volumetric fusion neutron source with yield of more than 1E18 n/s. The integration testing of nuclear system engineering could be performed.
As an important platform for fusion technology and safety research, HINEG can be used to carry out the neutron activation and irradiation testing not only for structural but also functional materials to assess the performance and reliability, such as structural materials for the blanket, neutron multipliers and ceramic breeders for tritium fuel production, suitable radiation resistant thermosets for the electrical insulation of the superconducting magnets, in-vessel conductor coils, liquid-metal coolants, etc. The fusion neutron irradiation testing is being conducted on China Low Activation Martensitic (CLAM) steel, which has been developed by INEST and selected as the primary candidate structural material for Chinese Helium Cooled Ceramic Breeder ITER Test Blanket Module

(CN HCCB TBM). Moreover, the performance of components under neutron irradiation can also be assessed on HINEG platform, such as the tritium breeding blanket and shielding blanket.

O1.B / 53

Experimental refutation of the deuterium permeability in vanadium, niobium and tantalum

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Unique gas retention and transport characteristics of group V elements (V, Nb, Ta) have long attracted a significant interest, in particular among the nuclear fusion community. The nominally high hydrogen isotope permeability and diffusion at the expected operational temperatures, together with the negative activation energy for the solubility present these materials as a promising choice for the fabrication of tritium recycling structures.
However, before seriously considering these materials, one should question the accuracy of the available data, given the remarkable lack of direct experimental measurements in support of the traditionally accepted transport properties of these materials. Furthermore, it must be considered that data have been mostly obtained by combining results obtained by different authors and methods.
The extensive literature review presented in this paper shows that existing experimental results not only contradict the semi-empirical values assumed for these materials but also present a broad dispersion.
In order to clarify this, deuterium permeability data for the three materials was obtained at the THERMOPERM facility at Ciemat (Madrid, Spain) in a relevant range of pressures and temperatures. Experimental difficulties together with the role of surface oxidation which may become a major issue for practical uses are also assessed.

O1.C / 60

High temperature microstructural stability of self-passivating W 10Cr-0.5Y alloy for blanket first wall application

Co-authors: Carmen García-Rosales [1]; Elisa Sal Broco [2]; Aida Calvo [2]; Iñigo Iturriza [2]; Nerea Ordás [2]; Jens Reiser [3]

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Tungsten is the main candidate material for the first wall (FW) armour of future fusion reactors. However, a loss of coolant accident with simultaneous air ingress into the vacuum vessel would lead to temperatures of the in-vessel components exceeding 1000°C, resulting in the formation of volatile and radioactive tungsten oxides. A way to prevent this important safety concern is the addition to tungsten of oxide-forming elements, which, in presence of oxygen at high temperatures, promote the formation of a self-passivating layer protecting tungsten from further oxidation.
A W-10Cr-0.5Y alloy produced by mechanical alloying and hot isostatic pressing (HIP) has been recently developed, exhibiting a strong reduction of oxidation rate compared to pure W and high mechanical strength. After HIP at 1250℃ it shows a two phase microstructure, according to the W-Cr phase diagram, with a nanocrystalline structure of the W-rich phase due to the presence of a Y2O3 nanoparticle dispersion inhibiting grain growth. A heat treatment after HIP at 1550℃ results in a one-phase material with average grain size of ⊠250 nm and coarsening of the Y2O3 particles to ⊠50 nm. This material exhibits a high thermal shock resistance, as demonstrated by tests at the JUDITH facility consisting of 1000 ELM-like pulses, where the material showed a comparable performance to pure W. Nevertheless, the microstructure is metastable and its thermal stability under

operational conditions has to be assessed.
In this work, results of thermal stability tests on heat treated W-10Cr-0.5Y alloy subjected to temperatures of 650, 700 and 1000℃ for times ranging from 50 to 3000 h are presented. After 100 h at 700℃ a slight growth of the Cr-rich phase is detected. After 100 h at 1000℃ a complete decomposition takes place with the formation of a uniform, fine-scale mixture of W- and Cr-rich phases, typical for spinodal decomposition.

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Neutron spectrum determination at the ITER material irradiation stations at JET

Co-authors: Lee Packer [1]; Paola Batistoni [2]; Steven Bradnam [3]; Bethany Colling [3]; Sean Conroy [4]; Zamir Ghani [3]; Mark Gilbert [3]; Slawomir Jednorog [5]; Ewa Laszynska [5]; Dieter Leichtle [6]; Igor Lengar [7]; Jerzy Mietelski [8]; Ryszard Misiak [8]; Mario Pillon ; Sergei Popovichev [3]; Vladimir Radulovic [7]; Ion Stamatelatos [9]; Theodora Vasilopoulou [9]; Anna Wójcik-Gargula [10]

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The experiments that are planned over the next few years at the Joint European Torus (JET), notably including a deuterium-tritium (DT) experimental phase, are expected to produce large neutron yields, up to 1.7E21 neutrons. The scientific objectives of the experiments are linked with a technology programme, WPJET3, to deliver the maximum scientific and technological return from those operations, with particular emphasis on technology exploitation via the high neutron fluxes predicted in and around the JET machine. Importantly, the programme aims to extract experimental data relevant to the international effort to design, construct and operate ITER. The data expected to be retrieved under the JET experimental program will support, develop and improve the radiation transport and activation simulation capabilities via benchmarking and validation in relevant operational conditions. Such capabilities are important and are applied extensively to predict a wide range of nuclear phenomena and impacts associated with components and materials that will be used in ITER operations.

This paper reports the status of activities conducted as part of the ACT sub-project collaboration under WPJET3. The aim of the subproject is to take advantage of the significant 14 MeV neutron fluence expected during JET operations to irradiate samples of materials that will used in the manufacturing of main ITER tokamak components. The paper will provide analysis of the characterisation work at irradiation stations at JET performed in a previous deuterium campaign using dosimetry foil measurements, and give the status of irradiation experiments at JET that are ongoing in 2018 using real ITER materials. The experimental results are further used, together with calculated dosimetry foil response functions (Ti, Mn, Co, Ni, Y, Fe, Co, Sc, Ta) and spectrometry unfolding methodologies, to derive neutron spectrum information at irradiation positions, which are compared to those derived from neutron transport simulations.

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Multifunctional nanoceramic coatings for future generation nuclear systems

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Several breeding blanket concepts for the DEMO reactor employ the eutectic Pb–16Li as breeder material, namely Helium Cooled Lithium Lead (HCLL), Water Cooled Lithium Lead (WCLL) and Dual Coolant Lithium Lead (DCLL). These three concepts share, with different incidences, three major technological challenges: tritium containment, steel corrosion and magnetohydrodynamic drag. Here, we describe the ongoing work on multifunctional Al2O3 nanoceramic coatings grown by Pulsed Laser Deposition (PLD) and Atomic Layer Deposition (ALD) on T91 steel. In fact, these two techniques are complementary from the manufacturing point of view since the first can produce relatively thick (up to 10s of µms) high performance coatings, while the latter is capable of coating complex 3D objects with thin films (in the order of 100s of nms). Both coatings were tested as tritium permeation barriers with hydrogen at different temperature (from 350 to 650 ℃). Results collected in this way indicate an excellent behavior, with a permeation reduction factor (PRF) up to 10^5 for both PLD and ALD coatings. In the case of PLD grown Al2O3 coatings, these results have been shown to be maintained also in the case of deuterium under 2MeV electron irradiation. Moreover, the electrical conductivity of these dielectric coatings is shown to be extremely low even when subjected to irradiation. ALD coatings are being currently tested in these conditions. Finally, to evaluate the chemical compatibility of Al2O3 films in liquid eutectic Pb-16Li, PLD and ALD samples have been exposed to static corrosion tests up to 8000 hours. No corrosive attacks on the steel substrate are detected. In conclusion, alumina coatings deposited by PLD and ALD show great promise to tackle the major technological challenges associated to the BB concepts employing Pb-16Li as breeder materials.

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European Integrated Programme in support to ITER: Overview of JET and Medium Size Tokamak results

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Europe has elaborated a Roadmap to the realisation of fusion energy in which 'ITER is the key facility and its success is the most important overarching objective of the programme'. EUROfusion has seized the unique opportunity to develop an integrated programme on devices of different sizes, i.e. on EU Medium-Size Tokamaks (MSTs), and, on JET in order to provide a step-ladder approach for extrapolation to ITER. In addition, the ITER Organization has issued a detailed analysis of the risks to ITER operation and has identified the main R&D needs to mitigate those risks in the revised ITER research plan. In this context, this paper will provide an overview of the recent coordinated contributions of the EU programme to optimise ITER operation.
Disruptions are considered as the highest operational risk in the ITER Research Plan. The high priority physics studies on JET and MSTs consist of disruption prediction, avoidance, mitigation and associated modelling (including multi-machine run-away electrons model validation). A new shattered pellet injection system is being installed on JET to compare with massive gas injection and elucidate the differences in run-away electrons beam mitigation in view of impacting the design of the ITER disruption mitigation system. The JET and the MSTs programmes have concentrated on the preparation of ITER operating scenarios and on providing a physics basis for optimising fusion performance operation with metallic first wall materials. It is found on JET and ASDEX Upgrade, that plasma performance is significantly affected when plasma boundary conditions are modified which will affect the strategy to achieve the fusion performance in the coming JET deuterium-tritium campaign and

ITER QDT=10 main mission. In addition, preparation of the ITER non-active phase has been carried out in hydrogen on JET, and, in hydrogen and helium in the MSTs. The recent progress will be reviewed on plasma surface interaction with ITER first wall materials (e.g. beryllium and tungsten erosion/migration, helium and tungsten interaction), scaling of L to H mode power threshold, Scrape-Off-Layer physics, core and pedestal confinement with different hydrogen isotopes and helium, control of detached divertor scenarios using extrinsic impurity seeding, and options for ELMs control with pellets or with resonant magnetic perturbations, RMPs. ELMs control with RMPs has been established on ASDEX Upgrade in helium using methods developed for deuterium plasmas, addressing a ITER issue of the transferability of ELMs control methods.
To conclude, the success of ITER operation will also require integrating the experimental progress made in different fusion facilities through theory-based first principle and integrated modelling. The European Transport Simulator, ETS, for integrated modelling has undergone major development and has been benchmarked against TRANSP. The strategic movement towards the adoption of the ITER integrated modelling and analysis suite (IMAS) has been pursued by the continued support and validation of the IMAS infrastructure and extension of the EUROfusion experimental databases in IMAS.
This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.
1 See author list of "X. Litaudon et al., 2017 Nucl. Fusion 57 102001"
2 See author list of "H. Meyer et al., 2017 Nucl. Fusion 57 102014"

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JT-60SA Program Contribution to Fusion Energy

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JT-60SA is a highly-shaped large superconducting Tokamak under construction by EU and Japan. The mission of JT-60SA is to support ITER and to complement ITER towards DEMO by resolving key physics and engineering issues. Fabrication and installation of components of JT-60SA by EU-Japan Integrated Project Team are progressing on schedule towards the first plasma in Sep. 2020. On the Cryostat Base made by CIEMAT, the 340-gegree-part of the Vacuum Vessel (VV) has been placed and welded accurately by QST. By Feb. 2018, all 18 TF coils have been manufactured and cold-tested by ENEA and CEA and 14 TF coils have been assembled to the tokamak by QST. Manufacture of all 6 EF coils have been completed by QST. Commissioning of the cryogenic system was completed by CEA in Naka. High Temperature Superconducting current leads have been delivered by KIT. Commissioning of the power supply system (ENEA, RFX, CEA and QST) has also been implemented smoothly. The Cryostat Vessel Body has been delivered by CIEMAT.
The JT-60SA Research Plan (SARP) ver. 3.3 was issued in March 2016 by 378 co-authors (JA 165 (16 institutes), EU 213 (14 countries, 30 institutes): Using ITER- and DEMO-relevant plasma regimes and its sufficiently long discharge duration, JT-60SA enables studies on all the key physics issues for ITER and DEMO. From ~2030, the first wall will be changed from carbon to full tungsten-coated carbon. By integrating these studies, the project provides 'simultaneous & steady-state sustainment of the key plasma performances required for DEMO'. Such JT-60SA research activity includes consolidation of a "Plant Simulator". As for the first plasma and heating experiments, JT-60SA will start earlier than ITER by five years. Therefore, experiences and achievements in JT-60SA are expected to contribute to reliable operation of ITER.

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W7-X: Technology progress of the experimental campaign with divertor plasmas

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Wendelstein 7-X (W7-X), a fivefold symmetric stellarator located at the Max-Planck-Institute for Plasma Physics in Greifswald, Germany, was successfully taken in operation with short pulse limiter plasmas in 2015. Hereafter, ten symmetrically positioned un-cooled graphite divertors were installed, the plasma facing wall was refurbished with graphite tiles and various auxiliary systems and diagnostics were upgraded. The reinforcements allow for an increase of the energy input from 4 to 80 MJ.
The experimental campaign with island divertor plasmas was launched in August 2017. The main goal of this campaign is to demonstrate the capability of the un-cooled test divertor in high density and high power plasmas. In the island divertor concept, the divertor is positioned in front of the pumps (for the neutrals exhaust) and it only intersects the relatively cold resonant islands around the core plasma. In this way, convective heat loads are limited to 10 MW/m² and neutral pumping is effective, despite the fact that the divertor area represents only 10 % of the plasma facing surface.
Second goal of the experimental campaign is to prepare for the installation of the final water cooled divertor. The water cooled divertor is planned for 2020 and has to be installed with great care and high accuracy. It relies on the experience gained with the installation of the un-cooled divertor.
To improve the density in comparison to the first campaign with limiter plasmas, an injector of frozen hydrogen pellets was installed and a fast hydrogen gas inlet with piezo-valves mounted in cut-outs of the divertor were taken in operation. In addition, various diagnostics were added or enhanced before start of the divertor campaign.
The tests divertor campaign is split into two parts: During a short break a the middle of the experimental campaign, two so-called scraper elements are installed in front of their corresponding divertor to shield the sensitive edges of the divertor near the pumping gap. Aim of the scraper program is to compare the edge loads on the divertor with and without a scraper and to evaluate the impact of the scraper on the neutrals pumping efficiency. The scraper elements are monitored by additional diagnostics. The break was also used to take the first of two neutral beam injectors (NBI) into operation and to add or harden several diagnostics.

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P2.161 Tritium Removal system for the Experimental Pilot Plant for D-T separation located at Rm. Valcea

A pilot plant for tritium removal from tritiated water is in pre-operational stage at ICSI Ramnicu Valcea and is based on catalytic isotopic exchange (LPCE) between tritiated heavy water and hydrogen/deuterium followed by cryogenic distillation (CD) aiming to recover tritium. As any detritiation plant or tritium processing plant/laboratory, also the Expriemntal Pilot Plant for D-T separation (PESTD) from Rm. Valcea requires a tritium removal system (TRS) with the main aim to recover tritium from gases used during maintenance of the rigs prior to their discharge to environment. In the same time the TRS is continuously connected, during normal operation of the plant, to the tritiated water containing tanks within the plant for a pressure management inside, similar as for ITER-WDS. In the specific case of the PESTD, TRS was given a secondary active role and that is to operate in closed loop with the LPCE and CD, suppling the LPCE with a specific flow rate of tritiated water, at a specific T2 concentration by obtaining it from the product of the CD process. This secondary role has been given with the purpose of minimizing the on-site T2 inventory for various types of experiments with the plant.
This paper will present a general schematic of the processes involved and under implementation, together with some specifications of the equipment that make the systems.

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P2.071 Quantitative Deuterium detection by laser-induced-breakdown-spectroscopy in ITER relevant samples

The detection of retained nuclear fuel in plasma facing components (PFCs) is currently one of the critical issues for ITER because of the impact tritium can have on the machine operation and safety. Laser Induced Breakdown Spectroscopy (LIBS) is a promising technique providing both qualitative and quantitative composition of the chemical elements retained in PFCs: it does not require sample treatment or manipulation, it can work in-situ between fusion discharges or during maintenance periods, it is suitable for measurements at different residual pressures.
In this work, carried under WPPFC, the results of LIBS measurements at 100 mbar nitrogen pressure on samples simulating ITER PFCs are presented.
The LIBS system is composed of a Nd:YAG laser (wavelength 1064 nm, pulse width 10 ns, repetition rate 10 Hz, spot diameter 1 cm) focused by a quartz lens of 500 mm nominal focal length into a vacuum chamber through a 1064-AR coated window. Pulse energy was 150 mJ, spot diameter at the sample surface was 2 mm. Light from the laser-induced plasma was focused into an optical fibers bundle (200 um core diameter) and sent to a 550 mm monochromator (resolution 0.1 Å at 500 nm) for the spectral analysis.
Small Mo tiles coated with 3 μm thick W-Al-D mixed layer (W 80% - Al 15% - D 5%) were used as samples, to simulate tungsten tiles contaminated with tritium (represented by deuterium) co-deposited with Be (represented by aluminium) eroded from the first wall.
LIBS spectra showed clear W, Al and D lines. Quantitative estimation was performed by applying the Calibration Free (CF) analysis. CF does not require reference samples so it's particularly suitable for this purpose. CF results have been compared with the nominal concentration values and found in good agreement. The aerial D density was found to be about 6·1017 D/m2.

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P2.088 Influence of the flow imbalance among the cooling channel on the magnet performance

Cryogenic circuit for cooling of a superconducting magnet like tokamak has a lot of branch and has to be designed efficiently considering the conductance for cooling. The KSTAR PF cryogenic circuit has one hundred eight cooling paths for fourteen superconducting magnets and CS structure. The five cryogenic valves has been installed to provide the same mass flow rate to cooling channel of magnet. The individual flow imbalance test has been performed one of the manufacturing process and it was below 10 %. But, we knew that the flow imbalance increased during magnets operation by installed of 10 mass flow rate meters in front of magnets additionally. In this paper, this flow imbalance will be applied to the KSTAR PF circuit modeling, which has been developed using SUPERMAGNET code, and the effect will be compared with the experiment data.

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P2.165 On the effect of stiffening plates configuration on the DEMO Water Cooled Lithium Lead Breeding Blanket module thermo-mechanical behaviour

Within the framework of the pre-conceptual design of the EU-DEMO Breeding Blanket (BB) supported by EUROfusion action, the University of Palermo is involved, as ENEA linked third-party, in the development of the Water Cooled Lithium Lead (WCLL) BB concept.
Results of the research activities carried out have highlighted that changes in the proposed WCLL BB design have to be considered, especially as to liquid breeder circulation path within the BB module. In particular, the suppression of breeder manifolds in the back supporting structure has pushed the WCLL BB design team to find an alternative solution, potentially relying on the snake-like radial circulation of liquid breeder through horizontal cells delimited by toroidal-radial Stiffening Plates (SPs), alternatively opened in their front and back side, respectively.

Therefore, in view of the definition of a final WCLL BB module layout, a parametric campaign of analyses has been carried out at the University of Palermo to assess the impact of different SPs configurations on the module thermo-mechanical performances. To this purpose, attention has been focussed on the three horizontal cells located in the outboard module equatorial region and the thickness and pitch of both vertical and horizontal SPs, as well as the radial width of horizontal SPs openings have been considered as parameters to be investigated. A Python script has been set-up to generate the ~300 models reproducing the three horizontal cells considered, perform calculations and post-process results. The Over-Pressurization (OP) accidental scenario, reproducing an in-box LOCA and mainly characterized by a uniform pressure of 18.6 MPa acting onto all water and breeder wetted surfaces has been assumed as the reference loading scenario.
The study has been carried out following a theoretical-numerical approach based on the Finite Element Method (FEM) and adopting the Abaqus FEM code. Results obtained are herewith reported and critically discussed.

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P2.077 TF Coil Inter-Layer Joint for EUROfusion DEMO Tokamak

The Swiss Plasma Center (SPC) has developed a Toroidal Field (TF) layout for the EUROfusion DEMO tokamak, based on a reference baseline of 2015. Each TF coil winding pack consists of 12 single layers wound with Nb3Sn graded conductors, connected in series by inter-layer joints, which are embedded in the winding pack. The react-and-wind (R&W) manufacturing technique is foreseen for the TF coil winding and for the preparation of inter-layer joints. The development of inter-layer joint is being continued at SPC in 2018 in frame of R&D program for TF coil of EUROfusion DEMO tokamak.
The high-grade Nb3Sn conductor, operating at 63 kA and 12.4 T (Tcs >6.5 K) was tested at SPC, and afterwards was used for the preparation of TF coil inter-layer joint. The inter-layer joint is an "overlap-type" joint. The two ends of conductors are copper-plated by a plasma-spraying method and joined together by a diffusion bonding at applied pressure and temperature.
This paper describes the manufacture of inter-layer joint, the preparation of sample and the test results obtained at SPC in the SULTAN test facility.

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P2.174 Systems Engineering approach in support to the breeding blanket design

Nowadays the Systems Engineering (SE) methodology is strongly applied in several fields of engineering such as Chemical and Process Industries, Civil and Enterprise application as well as Service and Healthcare systems. Furthermore, the SE represents a powerful interdisciplinary mean to enable the realisation of complex systems taking into account the customer and Stakeholder´s needs by defining the main functions and the associated requirements.
Also in the fusion community, this theme is becoming increasingly pressing and the implementation of the SE approach, from the early stage of the fusion reactor design, is now a must. Indeed, within the framework of EUROfusion activities, the SE method has been selected for capturing the system and interface requirements and for their management and verification with particular focus to the Breeding Blanket (BB) System of the European Demonstration Fusion Power Reactor (DEMO).
Specifically, various levels of functions and requirements have been elicited and a sophistication of the BB SE model, set-up using the Systems Modeling Language (SySML), has been performed. Indeed, a dedicated BB SysML model is also an integrating part of the SE methodology in order to support architecture development and requirements trade-off studies.
This paper describes the advantages of applying a SE approach to the BB design considering, in

particular, the BB requirement development and management and the definition of the interfaces between the BB and the major interconnected systems, including Remote Maintenance, Balance of Plant, Vacuum Vessel attachment, Heating and Current Drive and Fuelling Lines Systems.
An effective application of the SE technique to the pre-conceptual design phase of the BB is also provided in this paper. The obtained results are herewith reported and critically discussed.

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P2.197 Spinel-nitride based radiation tolerant optical materials

Radiation induced lattice defects strongly affect functionality of optical components, which will play a substantial role in various diagnostic systems of future fusion reactors. It is widely recognized that spinel lattice of double oxides (e.g. MgAl2O4) demonstrates enhanced radiation tolerance. One can expect a higher radiation tolerance of single cation spinels because in this case accommodation of radiation induced defects by swapping the positions of cations between tetrahedral and octahedral sites does not change anything in the lattice. Such kind of compounds are spinel nitrides (M3N4, where M = Si, Ge, Sn) recently discovered by one of us [1]. These compounds appear to be highly efficient luminescence materials suitable for fabrication of the new generation of LEDs covering the entire spectral range of the visible light. In this work spinel nitride samples were synthesized by laser heating in a diamond anvil cell and also using a high-pressure multi-anvil method. Self-healing ability (recovery of radiation induced defect) of interphase boundaries are expected since they are working as sinks for interstitials and vacancies. Therefore, solid solutions of (SiGe)3N4 were investigated in this work as well. To get the data on the electronic band structure and structural defects energy- and time-resolved optical spectroscopy using synchrotron radiation sources and cathodoluminescence setups were used [2, 3]. Our results confirm the theoretical prediction that spinel nitrides can be considered as basic materials for very efficient light emitting diodes suitable for application in demanding conditions under mechanical, thermal and radiation stress.
[1] Zerr A. et al. 2006, Advanced Materials 18, 2933.
[2] Museur L. et al. 2016, Scientific Reports 6, 18523.
[3] Feldbach E. et al. 2016, Optical Materials 55, 164.

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P2.193 Synthesis densification and mechanical properties of nanometric tungsten for fusion applications

Recent works have shown that low grain sizes are favorable to improve ductility and machinability of tungsten, as well as the resistance to ablation and spallation, which are key properties for the use of this material in thermonuclear fusion environment [Reiser et al. Int. Journal of Refractory Metals and Hard Materials, 64 (2017) 261]. However, current production routes are not suitable for the fabrication of bulk nanostructured tungsten samples. We propose here a new methodology based on powder metallurgy. Powder synthesis is performed using the Self-propagating High-temperature Synthesis (SHS) process. SHS is based upon the reduction of tungsten trioxide with magnesium in excess, and using sodium chloride as a reaction moderator. Resulting powders show platelet-like grains as the main feature, below 1µm in their largest dimension and around 60nm in thickness. Densification is then performed using Spark Plasma Sintering (SPS) at temperatures of up to 2000℃, even though the densification seems to be complete below 1600℃. Relative densities of 99.99% have been obtained. While Scanning Electron Microscopy (SEM) reveals an apparent grain size in the micrometer range, Electron Back Scattered Diffraction (EBSD), which is sensitive to the grain crystallographic orientation, clearly indicates that these micrometric grains are in fact nanostructured (see figure). A global preferential crystallographic orientation is observed, although the surface an-

alyzed is only 60x60 µm2. Simultaneously, this nanostructure induces an increase in hardness to a value of 428 HV, much higher than microcrystalline tungsten (~327 HV). These first results confirms that the process presented in this paper go in the right direction e.g. bulk nanostructured tungsten material. Mechanical tests (such as compression test) are currently ongoing with different test samples. Plasma exposure experiments are also undertaken in order to evaluate how the new nanometric W material sustains plasma erosion. All these results will be presented in this contribution.

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P2.143 The welding deformation control of large complex heavy-load vacuum Vessel port hub

Tailor-welded blanks are used in the manufacture of CFETR (China Fusion Engineering Test Reactor) port hub.In order to obtain high manufacturing precision, CFETR port hub is welded into a whole by EBW which has features of higher precision smaller deformation then melt welding methods.According to the inherent strain theory, the inherent strain of EBW welding is calculated ,then the result is imported into software.Simulation of welding deformation of four kinds of port hub welding program without clamp is to determine the optimal welding sequence and fixture constraints,then do the simulation of welding deformation of four kinds of port hub welding program with clamp.The results show that port hub with a different welding sequence results in different welding deformation.Fixture constraint can effectively reduce welding deformation.The results provide a reference for the actual welding process of port hub.

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P2.147 Key Technology Research of Electron Beam Welding in CFETR Vacuum Vessel Collar

Chinese Fusion Engineering Testing Reactor(CFETR) is a super conducting magnet Tokamak, and the key component begun to be studied in advance. 1/8 full size vacuum vessel(VV) as a research project, which purpose is to fully grasp the key technology of molding, welding , non-destructive testing and measurement in the aspect of building large-scale vacuum, and accumulate experience for the formal construction of CFETR, the project was officially launched in 2015 by the Institute of Plasma Physics Chinese Academy of Sciences. In the process of manufacturing 1/8 full size of the VV, in order to reduce welding deformation, electron beam welding is performed in the VV collar, ASIPP builds a full set of electron beam welding system, the technology research work of electron beam welding has been carried out in VV collar, and has acquired initial results. This paper introduces the key technology and its development for the VV collar electron beam welding.

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P2.182 Scope of modification of the TRINITI site fuel cycle complex for the IGNITOR project tasks

Project of IGNITOR tokamak is one of main directions of scientific collaboration between Russia and Italy. Project is entering stage of technical design for location of the machine on TRINITI site in Troitsk, Moscow region.
The IGNITOR machine differs considerably from other machines based on tokamak concept by using

a super strong magnetic field (13 Tesla) and plasma current (11 MA). It will operate with short pulses (around 10 sec) and will not have tritium breeding blanket. Requirements to tritium fuel cycle are high because three tritium pulses a day are foreseen with total amount in the range 10 − 15 grams to be processed daily. Fuelling is foreseen to be based on gas injection. Control of plasma current is based on pellet injection. For vacuum conditioning of the plasma chamber by hot helium (at temperature around 510K) use of channels for cooling helium, normally operated at temperature of around 30K, is envisioned. It necessary to provide a full tritium processing cycle including storage and supply of tritium, plasma exhaust purification, separation of hydrogen isotopes, detritiation of gaseous and water streams, tritium recovery from plasma facing components, etc.
Creation of tritium fuel cycle is an important task in modification of the facility already existing in TRINITI. In 2017-2018 Russia carries out evaluation of its capability to support IGNITOR program and modifications needs. The target is development of full scale tritium fuel cycle applicable to support development of a fusion plant.

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P2.185 Nuclear analysis of the HCLL "advanced-plus" breeding blanket with single module segment structure

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In the framework of the Horizon H2020 program, the European consortium EUROfusion develops a conceptual design of a fusion power demonstrator (DEMO) with the capability of generating several hundred MW of net electricity and operating with a closed fuel-cycle by 2050. The breeding blanket is one of the key components of DEMO. It must ensure tritium self-sufficiency and heat removal functions. In this framework CEA, with the support of Wigner-CR and IPP-CR, is in charge for the design of the Helium Cooled Lithium Lead (HCLL) blanket. It uses the liquid lithium lead eutectic as tritium breeder and neutron multiplier and helium gas as coolant for both the Eurofer structure and the breeder. This paper presents the nuclear analysis carried out with the TRIPOLI-4® Monte-Carlo code for the HCLL "Advanced-Plus" breeding blanket design using the Single Module Segment (SMS) option in the European DEMO 2017 baseline. This baseline is characterised by a reduced outboard thickness for a better plasma stability. Compared to the former baseline outboard thickness is reduced by 30 cm. Previous study has quantified its impact on Tritium Breeding Ratio (TBR reduction up to -0.08). This major constraint lead to the need of SMS solution development with HCLL "Advanced-Plus" design to reduce the amount of steel in the breeding blanket for TBR improvement. HCLL "Advanced-Plus" design is the last HCLL reference design currently developed with the aim to improve the TBR. The main nuclear quantities: TBR, nuclear heating, neutron flux, displacement damage and helium production will be reported and discussed.

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P2.020 Maximum Likelihood Tomographic Method for the Analysis of Bolometric Measurements on JET

The accurate determination of the emitted radiation is an important element in the interpretation of Tokamak performance and in the design of experiments. The spatial distribution of the total emitted radiation is typically determined with quite sophisticated tomographic techniques. On JET, a new tomographic inversion method, based on the Maximum Likelihood, has been very recently developed for this purpose. Its main innovative aspect is the analytic determination of the confidence intervals in the emitted radiation levels. The method is computationally quite fast and can therefore be applied on a routine basis. Together with a systematic use of phantoms, it can have several very

interesting applications. In addition to allowing a specific optimisation of the tomography for the main plasma scenarios, it permits also a systematic evaluation of various instrumental issues such as the effect of the noise, the impact of missing channels and the influence of the geometry and of systematic errors in the reconstructions. These potentialities are shown with a systematic analysis of bolometric data collected on JET during the experiments with the ITER Like Wall.

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P2.160 The DEMO Helium Cooled Lithium Lead "Advanced-Plus" Breeding Blanket: design improvement and FEM studies

In the framework of the European "HORIZON 2020" research program, the EUROfusion Consortium develops a design of a fusion demonstrator (DEMO). CEA-Saclay, with the support of Wigner-CR and IPP-CR, is in charge of one of the four Breeding Blanket (BB) concepts investigated in Europe for DEMO: the Helium Cooled Lithium Lead (HCLL) BB. The BB directly surrounding the plasma is a major component ensuring tritium self-sufficiency, shielding against neutrons from D-T plasmas and heat extraction for electricity conversion.
During the last few years, the HCLL BB design has evolved from a TBM-like concept, based on the Test Blanket Module for ITER, to a so-called "Advanced-Plus" concept in order to increase Tritium Breeding Ratio (TBR). This last concept, which is now the reference design for the DEMO HCLL BB, is characterized by the absence of vertical stiffening plates allowing a reduction of steel amount. The thermo-hydraulic scheme and mechanical design of the HCLL BB has been adapted to fulfill design criteria. Although previous studies were encouraging, they brought out some weak points in the design.
For this reason, this paper is particularly focused on the evolution of the HCLL "Advanced-Plus" BB design in order to improve the behavior of the structure. Thermal and mechanical Finite Element Method (FEM) calculations performed with Cast3M-FEM-qualified code on the equatorial outboard module are presented and analysed with RCC-MRx nuclear design and construction code for #1 accidental condition in case of Loss of Coolant Accident (LOCA) and #2 normal steady state condition. The methodology, the assumptions as well as the results obtained in each case are reported and critically analysed, scrutinizing some open issues.
Furthermore, a comparative study is made between the "Advanced-Plus"-reference concept and the TBM-like design to identify the advantages and drawbacks of each one regarding thermo-mechanical results.

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P2.171 On the effects of the Double-Walled Tubes lay-out on the DEMO WCLL breeding blanket module thermal behaviour

Within the framework of the EUROfusion project activities concerning the EU-DEMO Breeding Blanket (BB), University of Palermo is long-time involved, in close cooperation with ENEA, on the design of the Water-Cooled Lithium Lead (WCLL) BB, which is currently under consideration to be adopted in the EU-DEMO reactor.
The WCLL BB concept foresees liquid Pb-15.7Li eutectic alloy as breeder and neutron multiplier, whereas pressurized subcooled water as coolant, with operative conditions typical of the PWR fission reactors (temperature in the range of 295-328 ℃ and pressure of 15.5 MPa). The cooling down of the BB is guaranteed by means of two separated cooling circuits: the one consisted in square channels housed within the complex of Side Walls and First Wall, where counter-current water flow takes place, and the one composed of a set of Double-Walled Tubes (DWTs) submerged in the Breeder Zone (BZ) and deputed to remove heat power therein generated.
A parametric thermal analysis has been carried out in order to optimize the DWTs lay-out within

the BZ of the WCLL BB outboard equatorial region ensuring that EUROFER components maximum temperature stays below the allowable value of 550 ℃ while the overall rise of coolant temperature complies with the design requirement of 33℃. In particular, the parametric study has been focussed on the assessment of the best DWTs configuration in terms of number, disposition and orientation (vertical and/or horizontal).
The analysis has been performed following a theoretical-numerical approach based on the Finite Element Method (FEM) and adopting the ABAQUS qualified commercial code.
In the paper, results obtained together with the main assumptions adopted in the parametric analysis are critically discussed and a new, sustainable DWTs configuration is proposed.

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P2.140 Upgraded concepts and design of an in vessel inspection system for fusion reactors

ENEA Frascati Laboratories has been involved since 2000 years in the development of a specific technology to afford the issue of the inspection of the internal vessel's surface of large fusion reactors characterized by hostile environmental conditions (high vacuum, temperature, high magnetic field, high fluxes of gamma and neutron radiations). The development took place in a series of intermediates steps under EFDA contracts and successively F4E Grants. An early prototype was developed to cope with the constraints of the JET machine then, starting from this experience, a second prototype was developed for the ITER fusion reactor. At the end of this phase the system has been redesigned keeping into account all the actual ITER environmental conditions and mechanical constraints and passing the conceptual design review by ITER organization, thus the design became the reference scheme for the successive industrial business case and procurement phase.
ENEA Frascati Laboratories continued in the development of this technology applying it to a wide spectra of applications like underwater survey in pool of fission reactors, submarine environment and in the fine arts survey.
More recently a complete revision of the design developed for ITER has been done, applying all the expertise gained in the years and in the different applications to overcame some limiting characteristics of the old design. The revision completely modified the scanning system part and added further functions in the optical part like embedded calibration systems and related calibration procedures in order to improve the performances, making them more insensitive to the environmental conditions. The new revision has been protected by an international patent.
The paper describes the main architecture and the characteristics of the new revision evidencing the improvements compared to the old design.

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P2.238 Experiment study of spectrum of air bubble flash lamp

As an import component of the amplifier of inertia confined fusion (ICF), the flash lamp pumping efficiency have a lot to do with the amplifier efficiency and. In this paper, a kind of flash lamp with a new design has been manufactured to acquire high performance and it has been proved to be able to acquire high performance than the tradition flash lamps. Comparisons between the new and the tradition flash lamps have been made in terms of spectra of different discharge stages. In the same drive circuit, when the current increases, the spectrum of the new structured lamp is much higher than that of traditional lamp and dominated by continuous spectrum. After the peak of current, the spectrum of the e new structured lamp is similar to the traditional lamp. The mechanisms of the improvements and the further optimization for the new structure are discussed.

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P2.001 Modelling and experimental validation of RFX-mod tokamak shaped discharges

Shaped Tokamak discharges with an insertable polarized electrode have been executed in RFX-mod to achieve H-mode regime. This was aimed at reproducing successful experiments of stable operation at q<2 by feedback stabilization of m=2, n=1 mode already performed with low and high-beta circular discharges. Equilibrium magnetic configurations with a wide range of plasma shapes have been experimentally produced and analysed by means of the linearized plasma response model CREATE-L. In order to provide a connection between computational tools and experiments, the purpose of this work was to develop a general procedure for computing linearized plasma response models through the CREATE-L code, in any kind of plasma regime with a high level of accuracy with respect to experimental data. This procedure involves the solution of a constrained non-linear minimization problem to estimate the CREATE-L free parameters by using an iterative scheme trying to minimize the discrepancy between the magnetic field experimental and simulated measurements provided by pick-up coils. Eleven experimental shots have been identified and considered in this study: all of them are Upper Single Null tokamak configurations spanning the whole range of poloidal beta achieved in the RFX-mod tokamak (low-β, intermediate-β, biased induced H-mode regime). A preliminary sensitivity analysis showed a non-negligible dependence of static equilibria on variations of the total plasma current with respect to the measurement provided by Rogowski coils. Thus, the total plasma current has been set as an additional degree of freedom in the minimization problem allowing it to assume values between the Rogowski measurement and the value of the discrete line integral of the poloidal magnetic field measured by the pick-up coils. In all cases under analysis, the iterative procedure showed that the most accurate equilibrium is obtained with a plasma current higher than the Rogowski measurement but lower than the pick-up coils line integral.

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P2.057 Tracking of neoclassical tearing modes in TCV using the electron cyclotron emission diagnostics in quasi-in-line configuration

An important goal of tokamak plasmas is the control of magneto-hydrodynamic (MHD) instabilities with low m, n (poloidal and toroidal mode numbers), which can influence the confinement time of energy and particles and possibly lead to plasma disruption. These instabilities, which appear as rotating magnetic islands, can be reduced or completely suppressed by a current driven by electron cyclotron waves (ECW) accurately located within the island. A fundamental requisite for this control technique is the ability to identify the island parameters (amplitude and radial position) and to vary accordingly the ECW deposition location.
Here we describe a control scheme of the steering mirror of the ECW source based on the real-time tracking of the island radial position realized using only the electron cyclotron emission (ECE) diagnostics in quasi-in-line configuration, i.e. with nearly anti-parallel propagation of the ECW and ECE beams.
The successful experimental proof of principle of this scheme, tested on the TCV tokamak, is here reported.

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P2.003 Final design of the JT-60SA pellet launching system for simultaneous density and ELM control

The key mission of the new tokamak JT-60SA is to conduct exploitations in view of ITER and to resolve key physics and engineering issues for DEMO. Its pellet launching system was designed to cover according requirements by providing a powerful and flexible tool for the control of density profile and ELM frequency. Therefore, the systems lay out had to be adapted for pellet injection via a guiding tube with an already pre-set geometry from the vessel inboard side. Modelling showed inboard launch is a must in order to achieve high fuelling efficiency; by analysing the potential pellet fuelling impact in all relevant plasma scenarios the optimized set of pellet parameters for fulfilling all the tasks requirements was elaborated. The feasibility of a mechanical centrifuge as pellet acceleration unit was studied. This approach would guarantee the precise pellet launch speed as needed to enable the best adaptation to the guiding tube transfer capabilities and accurate control of pellet frequency and particle flux as arriving in the plasma. While the appropriateness of the centrifuge principle has been already proven by several devices, the proposed version possesses a novel design employing several accelerator arms. Such, operation at heighten rates and with a more refined adjustment of pellet flux and frequency could be achieved. Moreover, it allows hosting several sources delivering different pellets and their simultaneous actuation. The appropriate control unit is designed to merge pellets from different sources and form a sequence for the simultaneous control of different basic plasma parameters as e.g. density and ELM frequency. This flexibility can also allow including additional applications like isotope fraction control or radiative power exhaust by the use of compound pellets. At present, the detailed engineering of all major components is in progress, aiming to provide specifications and get ready to prepare the procurement process.

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P2.023 Conceptual study of an ICRH system for T-15MD using Traveling Wave Antenna (TWA) sections

The low aspect ratio (a/R=2.2) D-shaped tokamak T-15MD with toroidal field of 2T on axis is currently under construction in the Kurchatov Institute [1, 2]. Ion-cyclotron resonance heating (ICRH) is considered as an important heating method for this device [1]. In addition, ion cyclotron current drive (ICCD) could contribute to sustain the non-inductive plasma current for long-pulse operation. To decrease the power density and associated high voltage a distributed antenna system is proposed for heating of the future fusion reactor with ICRH/ICCD. Among the different possible solutions a set of Travelling Wave Antenna (TWA) sections is considered as most promising [3, 4]. It optimizes coupling to the plasma, is load resilient and avoids a large Voltage Standing Wave Ratio in the feeding lines. A conceptual design of 2 superposed TWA sections with each 8 radiating straps, is made for T-15MD in view of operation e.g. at 60MHz for 2nd harmonic heating of H plasmas. This antenna, loaded by a simulated density profile of T-15MD, is modeled including its resonant ring feeding system. The ring feeding allows the recirculation of the non-radiated power and the termination of the TWA section on its iterative impedance. The paper describes the antenna design, the feeding ring tuning algorithm and expected performances of this antenna concept. The chosen geometry of the TWA sections is compatible with that of a future reactor and therefore this antenna in T-15MD represents also a testbed for DEMO.

[1] E.A. Azizov, et al., Status of project of engineering-physical tokamak, in: Proc.23rd IAEA Fusion Energy Conf., Daejon, Korea, 2010, Rep. FTP/P6-01.
[2] P.P. Khvostenko, et al., Fusion Eng. Design 124, 114-118 (2017)
[3] R. Ragona and A. Messiaen, EPJ Web Conf. 157 (2017) 03044.
[4] A. Messiaen and R. Ragona, EPJ Web Conf. 157 (2017) 03033.

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P2.025 Preliminary conceptual design of the DTT EC heating system

The Divertor Tokamak Test (DTT) facility has been proposed in the European roadmap to study solutions to mitigate the issue of power exhaust in conditions relevant for DEMO. The Italian DTT tokamak [1] (BT=6T, IP=5.5MA, R0=2.08m, a=0.65m and pulse duration of 90-100s) is being designed to allocate the optimal divertor magnetic configuration under reactor relevant power flow (PSEP/R>15 MW/m) in the scrape off layer. A mix of three heating systems (ECH, ICH and NNBI) will equip the machine to reach the target value of 45 MW at plasma. The present reference design considers a capability of 20-30 MW of EC power at plasma to support and assist different tasks such as bulk electron heating, non-inductive current drive, avoidance of impurities accumulation and MHD control. The gyrotron sources (1MW/170GHz/100s) will be based on the depressed collector technology with 50% efficiency and will exploit the experience gained in developing the solutions for ITER. Two Solid State High Voltage Power Supplies will feed the gyrotrons, the main one for the cathode (−55 kV, 50 A) and a second stage for the anode (35kV, 0.1A). A Transmission Line (TL) with 90% efficiency and 1 MW power handling is being considered and two solutions are studied: evacuated waveguide, as in ITER, and quasi-optical multiple-beam TL, in use at W7-X, are presented and discussed in terms of layout, dimensions and theoretical losses. The conceptual design of equatorial and upper launchers based on the front steering concept is being developed to reach the required deposition location. The EC wave absorption efficiency has been investigated and is presented here, considering a selection of injection angles and launching points with dedicated beam tracing calculations using the GRAY code [2].

[1] R. Albanese et al, Fusion Eng. Des. 122 (2017) 274
[2] D. Farina, Fusion Sci. Technol. 52 (2007) 154

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P2.005 Development of reactor relevant pellet launching system technology on ASDEX Upgrade

Controlling the plasma density in a future fusion reactor will be mainly attributed to pellet injection using a control algorithm based on a rather difficult density measurement. The underlying technology to capacitate the Pellet Launching System (PLS) for the requirements is challenging. The ASDEX Upgrade (AUG) PLS was retrofitted for this task, intensifying the integration into the Discharge Control System (DCS). Lessons learnt and their consequences for the design of a new system "from scratch" will be described.

The technology of the ice production process is discussed as well as the observed performance in view of isotope ratio accuracy and plasma fuelling performance, mimicking the reactor fuel (mixture D/T) by using a mixture of H/D. Since a fusion power plant will require a steady-state pellet source, the development of a control strategy in view of process control is mandatory. The procurement of a new pellet source was launched in order to enhance the existing centrifuge acceleration system. This contribution will show first considerations.

An innovative conceptual design will be presented comprising the potential to replace nowadays centrifuge systems. Present-day vacuum rotating feedthrough technology enables the installation of the motor for the accelerating arm on the atmosphere side avoiding a series of technical difficulties.

The existing PLS on AUG is in operation now for almost 30 years, no fall-back option is available right now. A conceptual design for a new PLS consisting of extruder, launcher and control systems (also for density control) with focus on reactor relevant technologies except tritium compatibility is under preparation.

Results presented here are complementary domestic activities to the EUROfusion WP TFV.

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P2.006 Passive control of runaway electron displacement by magnetic energy transfer in J-TEXT

During disruptions runaway electrons(REs) often drift from high field side to low field side in J-TEXT. It may damage plasma facing components when REs strike the first wall with high energies. In order to mitigate the damage, a novel approach called magnetic energy transfer(MET) based on the principle of electromagnetic coupling is presented in this paper. A set of extra coils with a high coupling coefficient with plasma is installed on the device, and a toroidal current can be induced in the coils during disruptions which can transfer the plasma poloidal magnetic energy out of vacuum vessel. Flowing in the same direction as the runaway current, the induced current can attract the runaway current to high field side, control the displacement of the RE beams and prolong runaway current plateau. Compared with vertical field control, a significant advantage of MET can be seen that the MET is passive control without power supply, while vertical field control needs a power supply and belong to active control. The J-TEXT experiment results show that the increase rate of RE beams' horizontal displacement can be obviously slowed. The runaway current plateau can be prolonged 4-5 ms and the control effect becomes better as the induced current in the MET coils increases. Thus the MET can control the displacement of RE beams effectively.

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P2.018 Wideband polarizers switches and waveguide for Electron Cyclotron transmission lines

Overmoded corrugated waveguide is used in high power microwave applications such as Electron Cyclotron Heating systems, where it is necessary to transmit high power at very low loss. In the primary propagating HE11 mode, corrugated waveguide is effective over a large frequency bandwidth. This operational flexibility becomes important in multi-frequency systems. For 50-mm diameter aluminium corrugated waveguide nominally designed for the ITER 170 GHz ECH system, the theoretical ohmic loss of the HE11 mode is around 0.3e-3 dB/m. In a possible dual-frequency system at ITER, the theoretical loss in the same waveguide only increases to a manageable 0.8e-3 dB/m at 104 GHz.

For ECH, it is also necessary to change the polarization state of the propagating wave. A pair of miter bends with different mirror groove depths is useful for this purpose. By rotating both mirrors, the pair can be used to generate any desired polarization state. The design is frequency sensitive, so it can be difficult to achieve polarizers that function properly over a wide bandwidth. General Atomics has built a pair of 63.5-mm waveguide polarizers for the TCV tokamak's ECH transmission line that are designed to operate at frequencies ranging from 82.6 to 118 GHz. In addition, polarizers have been designed for ITER's 50-mm diameter transmission line. A computer code that calculates both the required mirror rotation angles and the ohmic losses in the grooves predicts that these polarizers will function properly at both 170 GHz and 104 GHz.

General Atomics has recently fabricated a new class of waveguide switches with rotary actuators that have up to four waveguide outputs (or inputs). Such switches with three outputs have been supplied for 82.6-126 GHz transmission lines at TCV. These switches are inherently wideband.

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P2.002 W7-X NBI beam dump thermocouple measurements as safety interlock

In the upcoming operational phase OP1.2b of the Wendelstein 7-X stellarator in 2018 it is planned to have the Neutral Beam Injection (NBI) Heating System operational. Any un-absorbed heating power is dumped on the NBI beam dump graphite tiles that are cooled using CuCrZr-cooling structures. The Heat Shield Thermography (HST) system is present to prevent damage and overheating of the graphite tiles on these beam dumps. In addition, other interlocks (plasma density and ECRH stray

radiation interlocks among others) are present to prevent damage to in-vessel components in case of un-absorbed heating power. Since the HST as well as the other interlocks have a rather low safety integrity level (SIL) it was decided to use the available thermocouple measurements of the beam dumps as an additional interlock. Due to the relatively slow response of this type of measurement, the focus of this safety interlock lies on preventing major damage to the PV wall in case of a chain of malfunctions of HST, other heating interlocks and heating systems control. It is not implemented to fully prevent damage to the beam dump structure itself.

This paper describes the setup of the beam dump measurements, the upgrade of the electronics cabinets to SIL2-rated thermocouple measurement with alarm trip relais, the sequence for stopping the NBI beam and the analyses performed to determine the interlock alarm trip settings for operation in OP1.2b.

[1] P. McNeely et al., "Current status of the neutral beam heating system of W7-X", Fusion Engineering and Design, Volume 88, October 2013
[2] N. Rust et al., "W7-X neutral-beam-injection: Selection of the NBI source positions for experiment start-up", Fusion Engineering and Design, Volume 86, October 2011
[3] B. Mendelevitch et al., "Water-cooling system of the W7-X plasma facing components", Fusion Engineering and Design, Volume 98-99, October 2015

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P2.013 Ion Cyclotron Frequency Range Cold Magnetized Plasma Modelling in ANSYS HFSS

Achieving the plasma temperature expected for nuclear fusion requires external heating systems, such as dedicated Radio-Frequency antennas. Dimensions, power level and manufacturing cost which are at stake make it impossible to build scale-one mock-up during design and prototyping phases. For that reason, modelling the electromagnetic interactions between magnetized plasmas and Radio-Frequency antennas is mandatory for nuclear fusion research.

The modelling of the interactions between cold magnetized plasmas and Ion Cyclotron Resonance Heating (ICRH) antennas is generally assessed using specific codes. Antenna coupling codes often approximate the plasma to surface impedances described by 1D half infinite models. Depending of the mathematical approaches used by these codes, the modelled antennas can be described either in simplified 2D dimensions or in 3D complex geometries converted from CAD models. Such approaches do not allow one to work directly on realistic antenna model and to assess rapidly performance changes. Moreover, during the design phase, it is convenient to use the simulation to directly estimate the thermal and mechanical loads in the same software suite.

Thanks to the collaboration between the fusion community and ANSYS, the RF modelling software ANSYS HFSS supports non-homogeneous gyrotropic medium, which are used to describe magnetized cold plasma away from resonances. In this paper, ANSYS HFSS is used to model ICRH antennas coupling to cold magnetized plasma. A simplified antenna model is compared with the specific coupling code ANTITER for various plasma density profiles. It is found that the coupling performances can generally be reproduced in HFSS. Its finite-element implementation imposes inherent restrictions on the definition and boundaries of the plasma domain due to the gyrotropic media in which two propagation modes can co-exist. These restrictions are discussed and technical recipes are given to conduct satisfying antenna coupling calculations with ANSYS HFSS, which allow faster design phases and experimental comparisons.

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P2.021 Safety Systems in the ITER Neutral Beam Test Facility

The construction of SPIDER, the experimental device of the ITER Neutral Beam Test Facility (NBTF) devoted to the study of the ion source, is completed and SPIDER will soon start operation. The construction of MITICA, the full-size prototype of the ITER HNBI, is in progress. SPIDER and MITICA operation presents many possible hazards including radiation (D.Lgs.230/95–Cat.A), electrical, explosive gas, laser and fire risks. As these safety risks are typically not limited to a single experimental device, the NBTF team decided to use for people and environment safety an integrated approach, referred to as the NBTF Safety System, capable to deal globally with all risks. Safety in the NBTF is implemented first by design in the plant systems where possible; when safety by design is not sufficient, the safety system uses specific devices for active safety. As many different systems collaborate to guarantee the NBTF safety, a coordinating system, called Central Safety System (CSS), has been implemented to supervise all safety systems and provide the safety responsible officer an overall graphics representation of the NBTF safety. The CSS provides the safety coordination and presentation layer, but it is also required to implement directly safety instrumented functions. The paper will present the requirements of the Safety Systems deriving from the safety risk analysis carried out. It will also describe the design and implementation of the safety components with particular focus on the CSS that uses the Siemens SIMATIC S7 400FH PLC technology along with safe input/output remote nodes connected through double ring-topology Profibus redundant network. Complying with the IEC 61508/61511 standards, the system architecture uses the PROFISafe profile and the human machine interface is based on WinCC OA; special provisions are used for communication of safety-related information with the PLC.

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P2.014 Design and mock-up tests of the RING photoneutralizer concept for an efficient DEMO NBI

High energy (800 keV) Neutral Beam Injection (NBI) is one of the methods being considered to heat EU DEMO plasma [1]. A major issue of present NBI systems is the limited efficiency of the gas neutralizer (for ITER NBI ~55%), which impacts on the overall system efficiency. An attractive method, but still undemonstrated at full performances, is the photo-neutralization of the negative D-ion beam. In this process the energetic ions pass through an optical cavity where they impact on laser photons with a frequency chosen to maximize the neutralization cross section. The expected neutralization efficiency can be up to 70-90%.
A possible scheme for photoneutralization is named RING (Recirculation Injection by Nonlinear Gating) [2] where the second harmonic of a Nd:YAG laser is extracted and trapped within a non-resonant optical cavity. A mock-up of the optical cavity has been built in Consorzio RFX to study its performances in order to demonstrate the feasibility of the RING concept and its potentiality for a full-scale NBI photoneutralizer. The mock-up has been operated with a low repetition rate (10 Hz) Nd:YAG laser (λ=1064 nm). The optical alignment of the cavity appears not to be critical and first operations are aimed to achieve the 2nd harmonic generation (SHG) saturation regime. The measurements of the SHG efficiency using a Lithium Triborate (LBO) crystal confirm the non-linearity of the process with the crystal thickness. The optical properties of the recirculating light are described by measuring the beam envelope profile with a laser-triggered CCD camera and the trapped beam pulse decay time with a fast photodiode. Operations with the mock-up are fundamental to assess the optical cavity performance and estimate the loss channels affecting the beam recirculation.

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P2.015 A low power testbed for the WEST ICRF launchers and for the acceleration of their commissioning on plasma

This paper presents a milliwatt-range testbed that has been recently designed and manufactured for the RF characterization of the WEST ICRF launchers. The low power testbed is integrated into the TITAN test facility. This extends the capabilities of TITAN from testing at high voltage/high current parts of the launchers in vacuum, to the characterization of the launchers coupling capabilities, their impedance matching and their load-resilience at low power. These pre-qualification tests allow checking the launchers performances before their installation in the tokamak which in turn allows accelerating their commissioning on plasma.

The low power testbed is an RF load based on a mechanically rigid glass aquarium which can host various mixtures such as salty-water or Barium Titanate (BaTiO3) solutions. Indeed, these high relative permittivity isotropic and homogenous dielectrics can mimic qualitatively well the magneto-plasma at ICRF frequencies for the fast wave. Hence, the S-matrix of a launcher radiating into a plasma can be qualitatively well reproduced when the launcher is facing the RF load. The aquarium is furthermore installed on linear guiding system which allows changing the antenna-load distance in a 0-100 millimeter range with an about millimeter-precision. Sweeping the antenna-load distance can be used to assess the launcher's load-resilience.

As a first step, the aquarium has been filled up with a salty-water mixture. Hands-on experiments have been conducted with a Tore Supra antenna facing the low power testbed. The optimal salt concentration, maximizing the coupling resistance, has been experimentally assessed and this optimal salt concentration has been confirmed by modeling.

Finally, the paper discusses the low-power tests of a WEST ICRF launcher and some results are detailed. In particular the launcher is tunable in the specified frequency range (48-60MHz), and even beyond, and its load-resilience has been assessed through the low-power experiments.

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P2.007 Status and tasks for modernization of TRINITI site infrastructure for the Ignitor project

The project of tokamak Ignitor is one of the main themes of long-term scientific cooperation between the Russian Federation and the Italian Republic. Currently, negotiations on the development of technical design tokamak Ignitor with placement on the site of TRINITI (Moscow, Troitsk, Russia). The discussion on preparing of the Russian-Italian Inter-government agreement on realization of Ignirtor Project is in ongoing too.
Project Ignitor differs significantly from that currently under consideration of the projects of fusion reactors based on the tokamak. Tokamak Ignitor has a super strong magnetic field (13 T), in which the pulse discharge (about 10 sec) flows a powerful discharge current (11MA). The Ohmic heating is the main mechanism of ignition of the thermonuclear fusion reaction
The main purpose of this stage of research was to determine the current state of the infrastructure of power and engineering-physical complexes of TRINITI and the development of technical proposals for the modernization of these complexes for the task of the Ignitor Project implementing.
During the research the current state of equipment and communications of practically all the main elements of power and engineering infrastructure of the tokamak with a strong field (TSP) pilot-bench complex TRINITI was inspected, including stationary and pulsed power supply systems, vacuum, cryogenic, fuel systems and diagnostic complex. Based on the comparison of the dates on technical characteristics of the tokamak Ignitor and requirements to the livelihoods systems of the tokamak from the published sources were prepared technical proposals for the modernization of the experimental bench complex of TSF TRINITI under project objectives Ignitor. The work was carried out as a one of the important steps to prepare the Russian side to participate in the joint Russian-Italian development of the tokamak Ignitor technical project.

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P2.008 West Operation management Software Suite (WOSS): software assisting organization and follow-up of West operation

In order to operate a large research facility, one needs software tools assisting the organization and the operation follow-up. In the past, separated software tools were used on Tore Supra but for West, an integrated approach was chosen. The West Operation management Software Suite (WOSS) allows a streamline management of information from the planned program up to the realized experiments and the follow-up of the West technical status. The WOSS is split up in six modules: Timeline, Roster, Logbook, Physics summary, Systems status, Incidents.
The Roster module is implemented to schedule experimental campaign with daily experiment information, including links to the experiment description and the list of people who takes part in each experiment, while the Timeline module shows a multi-month calendar of experiments.
Follow-up of daily experiment is ensured by the Logbook module. The Logbook collects data and comments from responsible officers carrying out experiments on pulse by pulse basis. The Physics summary module, extracting data from logbooks, provides an overview of the current experimental session and is displayed in the control room.
The Incident and the Systems status modules support the monitoring of West sub-systems. All the operation incidents that happended on any West systems are dully collected in the Incident module as well as current and foreseen status of subsystems are published on the System Status module. Thanks to the integrated database concept, incidents and status data are shown in real time on the Logbooks.
The WOSS deal with a large quantity of information and is of precious help for the coordination support unit. In the article will be reported, how the WOSS is facilitating the coordination by grouping all information of each sub-system needed for the operation. Moreover, the article will present a report of the last campaign with an analysis of West sub-systems availability.

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P2.004 Three-dimensional disruption vertical stability and breakdown analysis of the Italian DTT device

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The reduction of heat loads of divertor target is one of the main challenges addressed by the European roadmap to the realisation of fusion energy. In particular, eight different missions have been identified overall, of which Mission 2 'Heat-exhaust systems' is specifically devoted to this goal. Recently, the Divertor Tokamak Test (DTT) facility [1] has been proposed with the aim of investigating alternative power exhaust solutions in view of DEMO, hence with the capability of including various divertor concepts (e.g. conventional, snowflake, super-X, double null, liquid limiter).
In this paper, we numerically analyse the recently revised up-down symmetric Italian DTT device, with the aim of evaluating the effects of deviation from axisymmetry on the plasma behaviour. This will be done resorting to CarMa0NL [2] (evolutionary equilibrium in presence of 3D conductors) and CARIDDI [3] (eddy currents equations in volumetric conductors) codes. Vertical stabilization, plasma breakdown and disruptions are analysed in this respect. Three-dimensional effects may be both detrimental (e.g. ports) and beneficial (e.g. first wall) on passive stability, so the overall effect is not obvious and must be carefully evaluated. On the other hand, having significant currents flowing in the first wall during fast transients poses significant challenges on the design, to take into account the aspects related to the field penetration in the breakdown phase and the electromagnetic loads on plasma facing components during disruptions.
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P2.010 REMAINING USEFUL LIFE ESTIMATION OF CRITICAL DIII-D SUBSYSTEMS

DIII-D plays a vital role in the development of the physics basis for fusion energy and the ITER design. Designed in the 1970's and built in the early 1980's, the system started operations in 1986 and has provided a reliable platform for fusion experiments for over 30 years. A hallmark of DIII-D operations has been its ability to adapt to the changing needs of the fusion research community and support a wide range of tokamak experiments for various stakeholder's.
An important factor in ensuring continued reliable operations is gaining a thorough understanding of the remaining useful life (RUL) of the DIII-D facility. Developing quantitative assessments of RUL requires a complex set of analyses to evaluate failure modes and effects, life models, operating history and present condition. General Atomics has conducted a pilot RUL assessment project to develop and refine a set of RUL estimation tools applicable to various DIII-D systems.
The pilot assessment categorized DIII-D systems as either repairable/replaceable or facility-life-critical and developed a multi-factor, quantitative assessment of RUL for a facility-life- critical system, the F8-coils. In contrast to repairable/replaceable systems, facility-life-critical systems perform essential DIII-D functions and are expected to last the life of the facility. The F8-coil RUL assessment included life estimates based on conductor and insulation fatigue, insulation ageing, conductor corrosion and material degradation due to radiation exposure.
The pilot F8-coil RUL assessment project showed that under typical operating conditions seen to date, the F8-coil may be expected to achieve a life of ~50 years or >106 shots. The pilot project further demonstrated that employing reliability engineering techniques coupled with modern analytical tools, quantitative estimates for RUL can be developed with reasonable investment. A description of the methodology employed and results will be presented.

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P2.009 Design and implementation of quasi-optical components for the upgrade of the TCV EC-system

The EC-system of the TCV tokamak is progressively being upgraded with the addition of two MW-class dual-frequency gyrotrons (84 and 126GHz/2s/1MW) being manufactured by Thales Electron Devices with the first gyrotron delivered to SPC at the end of 2017. In order to connect the two gyrotrons to the existing low field side and top launchers, new waveguide routing from gyrotron hall to TCV tokamak was designed and dedicated Matching Optics Units (MOU) have been developed. The internal optics of the system have been determined aiming at optimal coupling to the HE11 waveguides. The laws of quasioptics were used to find quadratic surfaces to shape an incoming Gaussian beam representative of the gyrotron output into a beam matching the proper field distribution at the waveguide entrance and with HE11 content compatible with the system requirements. A solution with one flat (movable) mirror and two shaping mirrors was found and characterized with the physical optics code GRASP. The resulting field distribution is then truncated and projected onto the HE11 component to evaluate the design solution (coupled power at the waveguide entrance >98.4% and HE11 content >96.8% for both frequencies). The model and the results of this analysis will be presented and compared to a model based on the Rayleigh-Sommerfeld scalar diffraction integral. GRASP was also used to evaluate preliminary misalignment effects in terms of coupled power to the waveguide and the first MOU design moved to the manufacturing phase. In parallel to the gyrotron integration and to extend the level of flexibility of the TCV EC-system, a modular closed divertor chamber is developed, requiring the X3 top-launcher to be redesigned. Preliminary antenna conceptual design studies including new curvature to cope with the requirements of modularity and flexibility will be presented.

This work is supported by the Ecole Polytechnique Fédérale de Lausanne (EPFL).

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P2.011 A conceptual system design study for an NBI beamline for the European DEMO

Neutral Beam Injection (NBI) is a robust, established heating and current drive method in fusion experiments. Among its strengths is high current drive efficiency that may pave the path for steady state operation of a tokamak reactor with an economically viable recirculating power fraction. For large tokamaks like ITER and DEMO the use of negative ions is mandatory due to the vanishing neutralisation efficiency of positive ions at the required approx. 1 MeV beam energy. While the design of ITER's NBI has long been finalized and the prototype is under construction, NBI for DEMO poses new challenges that go far beyond those of ITER and require new approaches. Besides the demand for very-long-pulse or continuous operation, compatibility with much higher neutron fluences, and higher availability, a significant increase in energy efficiency is required to render NBI a viable option. Currently, the wall plug efficiency is limited to about 27 % on ITER, mostly due to the low neutralisation efficiency in beam–gas collisions. New concepts such as photo-neutralisation promise to overcome this limitation, but their practical feasibility has not yet been demonstrated.

Within the work package Heating and Current Drive of EUROfusion's Power Plant Physics and Technology division a systematic and comprehensive system design study of a DEMO NBI beamline was launched this year in order to explore a broad range of options for each beamline component and mutual dependences. The components' design is mostly constrained by the chosen principle of the neutraliser, e.g. whether there is a need for residual ion energy recovery (ER) that may save the energy efficiency of the beamline even if the particle neutralisation efficiency is far from 100 %. We have chosen this as starting point and present first concepts for ER integration as well as our general approach to a comprehensive system design study.

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P2.012 Long-Pulse High-Power 170 GHz Absorbing Matched Load Tests and Developments

The future EC systems will consist of several gyrotrons sources providing MW-level millimeter wave power at a frequency around or above 170 GHz. The development of matched loads is necessary to test the new sources, the components for the transmission lines and the launchers, and must ensure high qualification for compatibility with the nuclear environment. The load low reflectivity and high power-handling capability are mandatory for testing as well as for an accurate power measurement capability. The development at IFP and LTC of several compact high-power prototypes during the last decade led to a refinement of the overall design, resulting in the present low-reflectivity vacuum-compatible loads. The activity on loads is supported by F4E in view of the development of the EU gyrotron for ITER and required high power tests at QST (Naka, Japan). Qualification is now supported by new tests at SPC (Lausanne, Switzerland) using the FALCON test-bed designed to test components and the EU gyrotron prototype for the EC system of ITER. These high power tests, performed on the first CW prototype (provided with 16+16 cooling channels in parallel) highlighted the need to improve the mechanical, vacuum and hydraulic design to reach the final goal of 1000s at ~1MW. Two long-pulse loads with new technological solutions have been built for the Japanese ITER gyrotron at QST, while a modified version of the load, designed to test an equivalent input power of 2MW at QST, has been provided for FALCON.

Acknowledgement
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P2.016 Physics of the Traveling Wave Array for DEMO with proof of principle on WEST

To decrease the power density and associated high voltage a distributed antenna system is proposed as ICRH system for the reactor. Among the different solutions a layout made from a set of TWA sections is considered as the most promising [1, 2]. It optimizes coupling to the plasma, is load resilient and avoids large values for the VSWR in the feeding lines. The total radiated power scales as the number of independently fed sections such that high reliability can be expected. A first electromagnetic design for DEMO consisting in 16 sections integrated in the breeding blanket is proposed. This system would have a power capability of 50MW in front of a low coupling reference plasma profile with only 15kV maximum strap voltage. The TWA concept for ICRH is innovative and very different from the traditional IC antennas. A test on WEST should provide a proof of principle of the validity of the TWA approach together with a comparison with the existing WEST IC antennas. The chosen geometry of the TWA section is compatible with one unit of the complete set for a future reactor.
The paper describes the underlying physics, the antenna design and expected performances from complete modeling of the antenna system including its resonant ring feeding layout. A resonant ring feeding is used to ensure that the total generator power is radiated in the plasma. A comparative modeling with the present WEST antennas will also be discussed.
The proposed extrapolation to DEMO is discussed together with the problem of its integration in the breeding blanket in an accompanying paper [3].
[1] R. Ragona and A. Messiaen, EPJ Web Conf. 157(2017)03044.
[2] A. Messiaen and R. Ragona, EPJ Web Conf. 157(2017)03033.
[3] J-M. Noterdaeme et al., “Progress with the Ion Cyclotron Range of Frequency System for DEMO”, this conference

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P2.026 Advanced NBI beam characterisation capabilities at the recently improved test facility BATMAN Upgrade

The test facility BATMAN was dedicated since its start in 1996 to the development of radio frequency driven negative hydrogen ion sources for ITER NBI with focus on formation and extraction of negative ions, technological developments and improved concepts. During 2017 the test facility has been upgraded in order to replace the former extraction system (which was derived from a positive ion accelerator from ASDEX Upgrade) with a new ITER-like extraction system comparable in size to an ITER beamlet group. In addition to the standard three grids extraction system a repeller electrode upstream of the grounded grid is installed which can be positively biased by 2 kV with respect to the grounded grid for reducing the amount of back-streaming positive ions and space charge blow up of the beam. For the magnetic filter field a current of up to 3 kA can be driven through the plasma grid, as well as permanent magnets embedded into a diagnostic flange or in an external magnet frame.
BATMAN Upgrade will focus now on the properties of a beam with an ITER relevant extraction system. One of the main diagnostics is beam emission spectroscopy with line of sight located at two positions from the grounded grid (26 cm and 130 cm) with spatial resolution in vertical direction. A newly developed tungsten wire calorimeter placed just 20 cm downstream of the grounded grid should provide quantitative measurements of individual beamlets, while the previous tungsten wire calorimeter at 2 m distance is still in use for qualitative beam profile diagnostic. Together with a beam dump calorimeter with a crosswise arrangement of thermocouples, beam divergence and uniformity can be studied. This is accompanied by modeling of the meniscus formation, the beam optics, and the beam transport up to the calorimeter with simulated BES diagnostics.

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P2.027 Equatorial electron cyclotron port plug neutronic analyses for the EU DEMO

Within the Power Plant Physics and Technology (PPPT) programme in the EUROfusion Consortium design activities are currently in progress for the development of a DEMOnstration Fusion Power Plant (DEMO). In this framework, the design of the machine and the integration of in-vessel components require neutronics analyses fundamental to verify the tritium self-sufficiency, the shielding requirements and the structural integrity of its components. In particular, the penetrations in the blanket and in equatorial port plug introduced by the electron cyclotron (EC) heating system, namely due to openings for the antenna waveguides, can lead to significant leakage of neutrons which may increase the nuclear loads of the superconducting toroidal field (TF) coils and material damages in the vacuum vessel (VV) beyond design limits as well as reduce the tritium breeding capability. In this study a three-dimensional MCNP calculations were conducted for the pre-conceptual designs of the EC port plugs and shielding optimisation were performed in order to ensure that the DEMO design limits are not exceeded.
Two configurations of the EC heating system both featuring 8 square 63.5 mm × 63.5 mm waveguides (WGs) arranged in two horizontal stacks or a single vertical stack were analysed and shielding solutions for EC port plug proposed. In the first configuration the main focus was on the nuclear heating of the TF coils and in the second case the neutron induced damage of the VV. Analyses were performed using a DEMO Water Cooled Lithium Lead (WCLL) with integrated EC configurations model of a half of the sector (i.e. 10° model) and relevant ones repeated with a full sector model (i.e. 20° model) to test the reliability of results. Additionally, the effect of the radiation on the WGs closest to the plasma was analysed as well as the impact on Tritium Breeding Ratio.

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P2.028 Mechanical design of the high powered helicon antenna and strip line feed in the DIII-D tokamak*

A high-powered "comb-line" helicon antenna for use within the DIII-D Tokamak is currently in design and fabrication at General Atomics. The antenna will drive current in high beta discharges using electromagnetic helicon waves. The high powered helicon antenna (HPHA) is expected to couple up to one MW of power into DIII-D plasmas at a frequency of 476 MHz. The antenna design includes 30 individual, inductively coupled modules fastened to six internally water-cooled Inconel back-plates. The end modules have special design features to provide RF stripline feed attachment points. Each back-plate is mounted to a pedestal, which is secured to the vessel wall via a combination of studs and welded brackets.

The mechanical design includes features to minimize and survive disruption forces, thermal loading from RF losses, vacuum vessel bake (350C), and installation considerations. Thermal stress design challenges include plasma heat loads, thermal ratcheting, and bake-out cycles that result in thermal growth differences between the strip line and the vessel. Splitting the back-plate into six sections each with a central pedestal support significantly reduces the disruption loads. A single strip line folded in half, fed near the fold provides a 180° phase shift between the strip line module connections for driving the two module straps. This looped mono RF coaxial stripline design was chosen for optimal properties in space limitations, ease of attachment, installation, structural rigidity and RF tuning ability. The strip lines are supported by a quarter-wavelength "stub", mid span, for support during thermal cycling and plasma disruption events. Multiphysics FEA analyses are performed to optimize the geometric shapes to meet the aforementioned design challenges.

A description of the mechanical design, fabrication, installation, and analyses of the HPHA and stripline feeds will be presented.

- Work supported in part by the US Department of Energy under DE-FC02-04ER54698 and DE-

AC02-09CH11466.

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P2.048 Upgraded gamma-ray diagnostics for JET DT campaigns

For the future JET deuterium-tritium (DT) campaigns different gamma diagnostics, in particular the JET Gamma-ray Camera (GC) and the JET Gamma-ray Spectrometer (GS), have been upgraded in the last few years. The main demands for new detectors to be used during DT campaigns are connected with expected high count rates of 0.5 Mcps and with a required energy resolution equal or better than 5% at 1.1 MeV. Upgraded detector systems are based on fast scintillators, LaBr3:Ce and CeBr3, coupled to the best suited photodetectors, e.g., a multi-pixel photon counter (MPPC), also known as a silicon photomultiplier or SiPM, and a photomultiplier tube (PMT) operating with an active voltage divider for GC and GS, respectively. Preliminary results obtained with new detectors, already installed at JET, will be presented and compared to those collected in laboratory conditions. Information on basic performance of alternative photodetectors will be included in addition. A short description of a dedicated software for analysis of spectra registered with scintillators will be given.

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P2.038 Concepts of the new ASDEX-Upgrade flight simulator

Discharge scenarios and control schemes in ASDEX Upgrade (AUG) are evolving more and more complex. Especially in physics investigations for ITER and DEMO sophisticated scenarios exploit the operational space. This increases the probability of design flaws or human errors in the pulse configuration, but also aggravates the potential damage in the failure case.
The ASDEX Upgrade Flight Simulator, will provide a fast and efficient simulation tool for testing and validating discharge scenarios, as well as control and monitoring functions, during their development and immediately prior to experimental pulse execution. This ensures, that the scenarios and settings are adequate to reach the experimental goals and that the margins to operational limits are sufficiently large also during the dynamic evolution of the discharge.
Simplified physics and plant system "control" models combined with a representation of the ASDEX Upgrade Discharge Control System (DCS) allow for fast simulation runs with reasonable prediction quality. In the simulation an event generator can trigger plasma instabilities, technical failures and external events to test the resilience of the designed pulse against unplanned incidents. The granularity of modelling shall be customisable, such that the simulator can also be used for detail investigations with elaborate physics at the cost of longer simulation time.
As a basis for implementation the ITER Plasma Control System Simulation Platform (PCSSP) has been chosen. The flight simulator, extends PCSSP with an ASTRA co-simulator for the ASDEX Upgrade tokamak model and with custom modules for its actuators, diagnostics and control system. Plugins will enable reading original AUG discharge programs and configuration files, as well as storing the results in the AUG shotfile database.

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P2.031 The dud detector: an empirically-based real-time algorithm to save neutrons and tritium during JET DTE2

Operations using deuterium-tritium mixtures are envisaged on JET in 2019-2020 (DTE2). Each plasma discharge will be a precious resource during this campaign, being both tritium and neutron budget limited. During DTE2 it will be mandatory to promptly detect and safely terminate those plasma discharges which do not achieve the expected target parameters, due to unsatisfactory plasma performances (i.e. low beta, low fusion power, non-stationarity) or unhealthy conditions that compromise the experiment goals (i.e. inadequate heating power, deleterious MHD, incorrect fuel mixture). A real-time detector of underperforming discharges (dud) algorithm has been developed for this purpose. The algorithm will calculate and monitor the time evolution of plasma performance indicators, which are then used to trigger alarms. Among these, the confinement scaling H98y,2 and the reactivity normalized to the plasma stored energy are the most promising indicators, since they can be easily, yet robustly, estimated based on the available real-time signals on JET. Alarm thresholds for such indicators have been empirically tuned over a wide database of advanced tokamak, baseline and hybrid plasmas. Notably, the robustness of such alarm thresholds will be tested in high heating power regimes and in presence of different isotope mixtures in upcoming JET campaigns. The performance of both detection and alarm generation has been characterized and documented. Coupling the dud detector to other real-time controllers (i.e. radiation peaking detector, isotope mixture and mode locking control) and to proper plasma termination strategies has been investigated in this work. Furthermore, a possible synergy with the real-time state observer RAPTOR code, which will provide a model-based expectation of the plasma state, is also discussed.
*See the author list of "X. Litaudon et al 2017 Nucl. Fusion 57 102001"

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P2.034 Benchmarking high performance ITER CODAC data archiving system

High-speed long pulse archiving systems are critically sensitive to the latencies produced by hardware and software along full archiving chain. Therefore, detailed studying of this phenomenon, estimating its impact on archiving process, correct selecting and commissioning of the hardware for archiving purpose, optimizing of archiving system configuration are indispensable steps to achieve stable loss-less data archiving.
The aim of this work is to present results of profiling tools development and investigation performed on ITER archiving system for DAN (Data Archiving Network), that includes data from diagnostics and fast control and with highest data archiving performance. Low invasive data collectors with millisecond resolution and ready to work in 10Gb data archiving network, have been developed as well as profiling data injectors. Processing and visualizing tools provides all required functionality for a conscientious study of disk/file system latencies and network performance with realistic estimations of minimal resources for loss-less archiving.
The work describes an efficient methodology of verifying suitability of given hardware for data archiving based on ITER CODAC technology, and also presents results of its application over a real data archiving infrastructure. They include observations of several phenomena, that were affecting real-time archiving and were detected on network and file storage system, and corrections on data archiving parameters to mitigate these phenomena.
Efficient methodology of verifying suitability of given hardware for DAN archiving purposes was developed.

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P2.040 Influence of pellet shielding on disruption mitigation in ITER

Adequate avoidance and mitigation of disruptions must be ensured if ITER is to meet its objective of

high performance burning plasma operation. A comprehensive disruption mitigation system (DMS) is being designed to ensure that thermal, electromagnetic and runaway electron (RE) loads are reduced to tolerable levels. The strategy relies on the injection of impurities/fuel using an array of shattered pellet injectors (SPI) situated in 3 of the upper port plugs and in one equatorial port. Concerning energy loads to plasma-facing components, a mixture of Ne/D2 shards is presently foreseen for mitigation of the plasma stored energy at the thermal quench (TQ) and to increase the density for avoidance of RE formation. A second injection of Ar will be made into the current quench plasma if RE beam development nevertheless occurs.
The TOKES code has for several years been used for the simulation of mitigation by massive injection of impurities. First simulations of SPI [1] with the code approximated the Ne pellet by a gas puff of artificially high density, equal to the average pellet debris density. However, this approximation is too simplistic: the injected pellet debris are evaporated and the vaporized Ne shields the shards from the hot plasma, inhibiting further vapourization and resulting in deeper penetration into the core.
This paper describes the results of TOKES simulations of SPI in ITER, similar to [1], but taking into account the plasma shielding of the pellet debris. TOKES shielding model captures main features, influencing Ne injection distribution. The simulations show that shielding of large pellets (2 kPa·m3), for which the amount of injected Ne is much larger than required for dissipation of the TQ thermal energy, have minor effect on the wall heat load. However, the minimum pellet size providing full dissipation is reduced by a factor of 2.

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

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P2.079 Design and experimental study on high current dc disconnector contact

The high current dc disconnector is the significant switch in ITER poloidal field converter power supply system. The high temperature rise in the contact area of disconnector is the key factor which limit the capacity of current carrying. In order to reduce the contact resistance and improve the capacity of current sharing and carrying, two new contact structures of circular contact and ring contact with spring to realize self-adaption are presented in this paper. Based on the built contact resistance test platform, the relationships between contact resistance and applied contact force, roughness of contact surface, and the shape parameters of contact, like curvature radius of circular contact, inner diameter and raised radius of ring contact, are introduced in detail. In addition, based on the ring contact structure, a nine-contact connected in parallel is set up to test the capacity of current sharing, and the results show that the non-uniform sharing coefficient is only 1.23, which demonstrates the good performance of the proposed structure.

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P2.042 Development of the new timing system for the ISTTOK Tokamak

The ISTTOK, a large aspect ratio fully ohmic tokamak operated at IPFN-IST is presently scientifically exploring an AC) regime, aiming to extend much longer pulses, up to one second plasma and around 40 current inversions.
The control of the earlier single-pulse plasma formation and sustaining was essentially deterministic using pre-programed delays on a set of timing channels generated within an in-house developed VME board. Due to limited register space, in different scales time delays the time resolution had to be adjusted and/or several channels chained together which implied low flexibility and a non-straightforward operator interface. Meanwhile the first AC operation campaign showed that the

current inversion phase is critical for the number of plasma semi-pulses that could attained, which pushed the development of a new ATCA/MARTe based real-time control system, capable to monitor the plasma and generate the optimized power supplies and other actuator references in a 50us control cycle. Following, also the timing and triggering system is now being upgraded with a dedicated custom hardware based on a high performing FPGA (Xilinx Kintex 7), which can be reconfigured on-the fly when receiving external events. In this contribution, we present the system architecture, main features, event transmission network, programing model, and some initial tests made in IST-TOK.

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P2.043 The timing system of the ITER full size neutral beam injector prototype

The ITER Neutral Beam Test Facility (NBTF) is currently under construction at Consorzio RFX, Padova, Italy. The NBTF includes two experimental devices: SPIDER for the study of the ion source and MITICA for the development of the full-size HNB prototype. Even if the two experiments share many architectural aspects in their Control and Data Acquisition System (CODAS), there are nevertheless some significant differences because MITICA must adhere to ITER's directives for plant control design. This paper describes the architecture of the MITICA timing system that represents one of the first examples of integration of IEEE1588 timing functions in fusion research medium-scale experiments.
Timing functions are required in MITICA CODAS to:
1) Synchronize data acquisition and assign a timestamp to each data sample;
2) Supervise the generation of events, trigger a reaction of the control system and/or provide enhanced data acquisition in a time window centered on the event occurrence time.
The IEEE1588 time synchronization protocol ensures sub-microsecond clock alignment among devices connected via a LAN. Therefore, it has the potential advantage of avoiding direct cabling among the involved actors, being all the synchronization messages exchanged over the LAN. A feature that IEEE1588 does not cover is the management of asynchronous events that may trigger some actions. Asynchronous events may trigger:
1) Enhanced data acquisition (e.g. with a higher sampling speed) during a time window centered on the trigger occurrence time;
2) Real-time response from the control system.
While the management of events triggering control actions is under the responsibility of the real-time control system, for events triggering enhanced data acquisition, the "lazy triggers" solution is adopted in MITICA. This solution allows events to be communicated via LAN, compensating the delay in network communication will be compensated by means of circular buffers at the ADC side.

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P2.051 Hydrogen isotope ratios measurements by Penning gauge spectroscopy of molecular Fulcher-α band

ITER and DEMO will use a optimum 50%:50% deuterium-tritium gas mixture as fuel. The fusion reaction rates depend on the hydrogen isotope ratios, therefore, these ratios are important to monitor both in the confined fusion plasma itself and in the pump ducts. Penning gauge spectroscopy of Balmer-α lines of hydrogen isotopes is widely used in present-day experiments to determine the hydrogen isotope ratios and the partial pressures in the pump duct. Such a diagnostic system is also foreseen in ITER.
The Balmer-α line isotopic shifts are very small <0.18 nm and the lines partially overlap because of

the presence of energetic atoms produced by molecular dissociation. The molecular ro-vibrational emission band of each hydrogen isotopomer consists of many narrow molecular lines, covering a wide wavelength span and have unique signature. The drawback is the low line intensities in comparison with atomic hydrogen lines from the most prominent Balmer series.
To investigate the capability of the hydrogen molecular spectroscopy to improve the hydrogen isotopic ratio determination, an Alcatel-type Penning gauge equipped with the optical window and the collecting optics was coupled by the optical fibre to the Echelle spectrometer having 370-680 nm spectral range and the spectral resolving power above 20000. The intensities of both atomic Balmer-α lines and molecular Fulcher-α bands were measured for gas mixture pressures in the range of 1E-6-1E-3 mbar. Different mixtures of H2 and D2 gases for given base vacuum pressure were produced by varying of both hydrogen and deuterium gas flows. The total intensities of the Fulcher-α molecular bands were determined by using the measured rotational and vibrational population temperatures. The comparison of isotopic ratios measured both by Balmer-α and by Fulcher-α bands spectroscopy will be presented. The capability of the molecular spectroscopy to determine the hydrogen isotopic ratio with high precision will be discussed.

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P2.045 Upgrade of Thomson scattering system on VEST

Upgrade of the Thomson scattering (TS) system in Versatile Experiment Spherical Torus (VEST) is planned for measuring the electron temperature and density with higher reliability and higher time resolution. The existing TS system has difficulties on measuring single plasma discharge, since it uses a laser with energy of 0.65 J and repetition rate of 10 Hz, while the pulse duration of the plasma is ~20 ms. Recently, additional heating sources such as neutral beam injection and electron cyclotron heating are installed, thus, the upgrade of the TS system has been required to provide the time evolutions of the electron temperature and density for heating efficiency analysis. The upgrade is mainly related with two parts. First, a new laser is utilized for increasing both the signal-to-noise ratio (SNR) and the time resolution. The laser generates 10 pulses with 1 ms time interval in every 2 s, and each pulse has the energy of 2 J. And a switched capacitor type digitizer with 32 channels and 5 GS/s is newly employed for storing a number of pulse signals. Second, the optical fibers for transferring the collected TS photons are improved as well as the laser. Although the collection solid angle and the numerical aperture between the collection lens and the optical fiber are well matched each other, it uses only half of the maximum etendue of the polychromator. Therefore, the fiber bundle is designed to optimize the optical properties among the lens, fiber, and the polychromator. Compared to the current system, replacement of the laser, modification of laser injection optics, and fiber bundles are expected to improve the efficiency more than 6 times. Furthermore, in the temporal point of view, it decreases the number of required plasma discharges to be a tenth.

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P2.046 Flexible vacuum vessel bolometer camera design in ITER to adapt to the final position of the gaps between blanket modules

Bolometer cameras in ITER will be mounted in Port Plugs, on Divertor Cassettes and on the Vacuum Vessel (VV) wall behind Blanket Modules (BMs). For the first assembly phase the platform (Cable fixations and the lower part of the internal signal chain) of VV cameras has to be delivered to fix the signal cables and protect their termination. During First Plasma, the as-built magnetic axis and magnetic flux surface will be measured and the BMs fixation will be adjusted to align them to the shape of the magnetic flux surfaces. Movements of the BMs of up to 15 mm in toroidal and vertical direction and up to 25 mm in radial direction are foreseen. Accordingly, VV cameras need

adjustments, too, to assure a proper view of the plasma.
Calculations have been performed to define the impact of the possible BM movements onto the bolometer viewing cones. The toroidal and poloidal movements can be followed by shifting the collimator and the sensor along with the BMs. The sensor temperature needs to be kept below 350 ℃. Thus, optimal thermal management of the cameras and an efficient thermal flow path to the VV wall is required. Therefore, the radial movement of BMs was transferred to an additional poloidal shift of the detector and collimator. The resulting camera design with its complex space envelope will be presented. It provides an alignment flexibility of the cameras of +/- 31 mm poloidally and +/- 15 mm toroidally. Because of the short period between magnetic measurements and second assembly phase, the number of customizable parts is kept as low as possible, which means in practice that only a part of the upper camera body and few elements of the internal signal chain are designed to be customizable. All other elements can be manufactured in advance.

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P2.050 Testing of the optical chain mock up of H-alpha and Visible Spectroscopy for ITER

H-alpha and Visible Spectroscopy is one of the ITER first-plasma diagnostics providing full poloidal coverage of plasma scrape-off layer near the first wall. There are two poloidal-view channels in EPP11, one tangential-view channel in EPP12, and one divertor-view channel in UPP02. At the moment, the final design phase is ongoing, requiring proper testing of design solutions to identify the realistic optical parameters and the problems related to the manufacturing, alignment, calibration and stability issues. A full-scale mockup of the whole optical chain has been built for this purpose. It comprises the most important units starting from the in-vessel optical front-end (First Mirror Unit) to the Interspace relay optics (Long-Focus Spectral Telescope), and to the Optical Bench Assembly located in the Port Cell, where the collected light is coupled to the imaging cameras and to the complex optical fiber bundle transmitting it to the various detectors and spectrometers to be located in the ITER Diagnostic Building. The detailed optical design, analyses, benchmarking and test results are reported with a main focus on the alignment/calibration issues, and on the spatial resolution measured. The measurements have been performed in the wide spectral range 450-700 nm, which is required for the monitoring of the H-alpha, Beryllium and other impurity emission profiles with probable local peaks considered as the most important ITER safety-relevant task for the passive visible spectroscopy diagnostics.

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P2.051 Development of Thomson scattering system for VEST

Thomson scattering (TS) system is developed to measure the electron temperature and density of Versatile Experiment Spherical Torus (VEST). Since it is the key diagnostics for measuring the local electron properties of the core plasma, each part of the system is carefully designed to provide a reliable measurement result. Besides, as additional heating devices such as neutral beam injection and electron cyclotron heating become available, the importance of the TS system has been emphasized to evaluate their effect. However, for VEST, measuring TS photons is especially challenging, because a low power of the laser and band pass filters of low efficiencies considerably decrease the number of scattering photons. In addition, the scattering signal and the stray light signal are detected in relatively short time interval due to the small chamber size. It means that decomposition of both signals is more difficult than conventional tokamaks. For successful measurement in such conditions, we have tried to minimize the noises such as stray light and high frequency interference from the pulse laser as well as the random background noise to obtain the enough signal-to-noise ratio. The time evolution of the electron temperature and density can be measured by shot-to-shot experiments

with the extra efforts for synchronization and optimization of the polychromator.

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P2.080 Superconducting properties of Nb thin films prepared on different Substrates

Nb thin films on different substrates have been prepared by DC magnetron sputtering for capacitor application. The influences of substrates on film morphology, crystallographic structure and characters have been investigated. Superconducting transition temperatures of the films have been measured and relationship between the transition temperatures and deposition conditions have been studied. The superconducting transition temperature (Tc) is found to decrease monotonically with the increase of the lattice parameter (a) irrespective of its thickness and grain size.

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P2.019 Experimental experience and improvement of NIO1 negative ion sources.

The ion source NIO1 (Negative Ion Optimization 1) is a versatile multiaperture H- source capable of continuous regime operation, with the plasma generated by a 2 MHz/2.5 kW radiofrequency (rf) power supply and nine beamlet extraction. It aims to partly reproduce the conditions of the much larger ion sources built or in construction for neutral beam injectors of fusion devices in a compact and modular ion source, where effects of individual source components can be rapidly verified and compared to simulation code results.
Several modifications of the magnetic configuration (both inside the ion source and for the embedded magnets inside the accelerator grids) were investigated, with material ranging from hard ferrite to SmCo to NdFeB; evidence of the effect of the accelerator fringe field inside the ion source is discussed. Saturation of filter field beneficial effect at large (>100 G) amplitudes is discussed, as well as the advantages of crossed field operation.
The radiofrequency system takes full advantage of the frequency tuning amplifier capability (+/-150 kHz installed, 0/+20 kHz used), and it is worth noting that no adjustment of matching box capacitors was necessary in the last 3 years.
Major diagnostic systems (including CCD cameras and BES measurement) have been integrated with the acquisition system along with the date measured by several power supplies (including the high voltage supplies, the rf generator and bias supplies, the pressure measurements and the beam currents).

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P2.058 Design Status of the ITER Core CXRS Diagnostic Setup

The Charge Exchange Recombination Spectroscopy diagnostic system on the ITER plasma core (CXRS core) will provide spatially resolved measurement of ITER plasma parameters. The optical front-end is located in upper port 3 and the collected light in the wavelength range of 460 nm to 665 nm is routed to spectrometers housed in the tritium building.
Continuing the efforts described in [1] of designing a better layout for the free-space optical chain, a new layout was selected. With the updated optical system in the port plug, cell and interspace,

the considerations which triggered the investigation, namely changes in the generic design of the upper port plug hosting the in-port diagnostic components and the concerns regarding the system lifetime as well as internal and external tolerances could be addressed. The new layout was chosen and optimised based on a number of additional criteria, including optical performance, radiation shielding [2], maintainability and robustness.
The main changes at the system level include the addition of a free-space optical chain to relocate the coupling of the light into the optical fibres to the port cell, avoiding significant irradiation of the fibre bundle. As a new sub-system, a line-of-sight finder based on the back-illumination and imaging of apertures at pupils and masks at images was added. With the line-of-sight finder, the absence of an object with detectable edges is addressed, enabling the determination of deviations within the optical chain in between and possibly during plasma shots.
Where feasible, existing solutions for sub-systems such as the shutter were adapted to the changed layout. Where not successful and for new sub-systems, a design is proposed.
The selection process of the optical system and the system-level consideration for the different sub-systems are discussed. The implementation of the main CXRS core sub-systems is detailed.

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P2.068 Upgrade of scattered light collection systems for KSTAR Thomson scattering diagnostic

Thomson scattering diagnostic system consists of a laser, a collection system, spectroscopy and a digitizer section. Recently, KSTAR Thomson scattering system has found some problems in collection system. The first problem is that the light transmission of the lens glass drops to less than 10% due to the browning of the lens due to the neutron, and when the plasma disruption occurs, the impact is transferred to the collection lens as it is, causing the Thomson scattering measurement problem. In order to solve these problems, we newly designed the core and edge lens of plasma and made a new lens using this design. This new lens has confirmed through simulation that vignetting and modulation transfer function (MTF) are improved compared to previous lenses[1]. Also, the collection lens installed in the cassette hanging on the cryostat is very vulnerable to vibration. To prevent such vibration errors, the design of the lens support has been changed so that the entire lens is floated without touching inside the cassette. This research mainly explains newly designed lenses and briefly describes lens supports.

Reference
[1] S. Oh, and J.H. Lee, Review of Scientific Instruments, 81, 10D504 (2010)

P2 / 633

P2.061 The highest spatial resolution infrared thermography on ITER-like tungsten monoblocs in WEST Tokamak

A new Infrared diagnostic has been developed by IRFM and installed in the WEST tokamak to measure surface temperature of the actively cooled W-monoblocs components as foreseen for the ITER Divertor units, with a very high spatial resolution of 100µm.

The goals are to investigate the effects of the shaping of these components on the heat load deposition pattern, the evolution of damages specifically introduced in WEST, the behavior of the leading edges regarding the assembling tolerances between adjacent monoblocs, and finally to help in the specification for the protection of ITER divertor.

In WEST, each Plasma Facing Unit is composed of 35 W-monoblocs of individual surface of 28x12mm. To analyze heat load pattern and phenomena on such tiny surfaces, the leading edges and in the

narrow gaps between monoblocs (400-500µm), a 100µm spatial resolution is required. Then, a Very High spatial Resolution (VHR) infrared diagnostic has been specially developed at IRFM.

The conceptual design of the endoscope is inspired by space industry. The endoscope structure is light, compact and "flexible-rigid" to endure transient events such disruptions. The VHR is made of 6 mirrors and 4 lenses to achieve the spatial resolution with 40% of encircled energy. The "home-made" WEST infrared camera is a 640*512 pixels matrix. The field of view is a 64*51mm rectangle, which is remotely moved in a 30x40 cm observable area on the divertor (motion of the 2 first mirrors). The VHR operates at 1.7µm wavelength to take advantage of the dynamic of the signal for the temperature range (300 to 3600℃).

The VHR infrared diagnostic is now operational above the divertor sector made of actively cooled W-monoblocs and graphite inertial components with W coating.
A description of the diagnostic will be presented as well as first surface temperature measurements obtained in WEST.

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P2.062 Validation of neutron generator emission rate estimated based on activation measurements during in-vessel calibration of JET neutron detectors

To determine the power produced in a fusion device, an accurate estimation of the neutron yield is necessary. It is a fundamental operational quantity and, being linked to plasma performance parameters, it is an important measure of fusion success. The neutron yield is also needed to support the operational safety case and is the prime input to operational and maintenance doses.
A second Deuterium-Tritium Experimental Campaign at JET is planned for 2020; in which up to 1.7·1021, 14.1 MeV neutrons will be produced. They will be measured by two neutron yield monitoring systems installed at JET: fission chambers (KN1) and activation system (KN2). These systems were absolutely calibrated in 2017 using a characterized ING-17 14 MeV neutron generator (NG) as the calibration neutron source. The neutron emission rate of the NG was measured based on monitoring foil activation measurements. For the NG characterization campaign, the absolute neutron emission rate of the NG was estimated with a total uncertainty equal to ±4.0%, while the neutron emission rate during the in-vessel calibration was assessed with an uncertainty ranging from 4.7% for Nb to 6.9% for Al.
An independent analysis of the activation measurements used to determine the neutron emission rate of the NG is presented in this paper. The neutron emission rates of NG from monitoring activation results were obtained using a detailed MCNP model of the NG and its neutron source properties. In the second case they were calculated using the FISPACT-II code with neutron flux spectra calculated by MCNP code. Calculated values of neutron emission rate were compared to values estimated based on the activation measurements. In the case of a NG characterization campaign the discrepancy between calculated and measured values of neutron emission rate was 3%. This discrepancy was lower for the in-vessel calibration, in the region of 1%.

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P2.063 Constraints on conceptual design of diagnostics for the high magnetic field COMPASS-U tokamak with hot walls

COMPASS-U [Panek et al. Fus. Eng. Des. 123 (2017) 11-16], a high magnetic field tokamak with hot walls, will be designed and built at IPP Prague replacing the currently operated COMPASS tokamak. Unique features of this new device bring noticeable constraints and requirements, which make

the development of necessary plasma diagnostics highly demanding. In the contribution, main expected constraints influencing a conceptual design of individual diagnostic tools for COMPASS-U will be reviewed and possible solutions will be introduced. Among the most important features of COMPASS-U leading to consideration or development of a new diagnostic design we can name: a high temperature of the tokamak walls about 300℃ to operate in a highly recycling regime; a high plasma density expected to be above 10^20 m^-3, which is characteristic for high magnetic field tokamaks; a high heat flux density (perpendicular to divertor targets) at the outer strike-point in the range 15-20 MW/m^2 caused by a short power decay length and a strong auxiliary plasma heating; the presence of 16 toroidal field coils and vertically symmetric poloidal field coils; and finally, the proposed future use of the liquid metal divertor. As a consequence, the diagnostic designs will require dedicated solutions for all in-vessel cabling, detectors and optical elements. We also show how divertor plasma opacity and geometrical and spatial issues connected with diagnostic elements at the high magnetic field side or with required tangential views are taken into account as well as an optimization of shapes of probe diagnostics (rail probe concept). Last but not least, liquid metals proposed to be used in a divertor introduce a question of the material compatibility, mainly at elevated temperatures, and their transport and redeposition on in-vessel components, including those important for optical diagnostics.

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P2.076 Electrical performance characterization of helium inlet of the ITER PF6 coil double pancakes

The helium inlet is one of the most important components of ITER Poloidal Field (PF) coils. The insulation structure of helium inlet is critical to provide sufficient electrical and mechanical properties in practical application. In this paper, an ITER PF6 coil double pancake helium inlet trial mock-up was designed and manufactured by simulating the actual manufacturing process. A thermal cycling test on the sample was carried out from 290 to 77 K, and before and after the thermal cycling, the turn-to-turn DC HV tests and Paschen test, also the final breakdown DC test after the thermal cycling test, were carried out. The test results satisfied the requirement of ITER. It is therefore verified that the PF6 coil double pancake helium inlet insulation has good electrical properties, and can be applied to the formal production of the PF6 coil double pancakes.

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P2.066 High performance image acquisition and processing system for optical boundary reconstruction on EAST

Tangentially viewed images of plasma from high-speed cameras are intuitive and reliable data which could be used to reconstruct the plasma boundary. A new image acquisition and processing system has been developed for optical boundary reconstruction on the Experimental Advanced Superconducting Tokamak (EAST). The head section of the optical system is an imaging lens, followed by fiber optic image bundles. In the end section, a relay coupling lens is used to connect to the sensor in the high-speed camera. The spatial resolution is < 5 mm for the designed field depth of 2200 mm, while the frame rate could be up to 10,000. In order to reconstruct the plasma boundary independently, the optical system has been calibrated using feature points on the tokamak before the experimental campaign began. The average error of calibration is less than 1 cm. The total acquisition and processing time for each image is less than 200 μs with current software and hardware. The optical boundary reconstructions of diverted plasma discharges are presented, showing good agreement compared with magnetic equilibrium reconstruction. As a potential diagnostic tool, the application of the high performance image acquisition and processing system is also discussed. It has the potential to be used in real-time plasma shape control and investigating of transient processes, such as

ELMy eruptions in H-mode, on EAST.

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P2.067 A clustering algorithm for scintillator signals applied to neutron and gamma patterns identification.

In many nuclear applications, sensors are widely used in order to detect high energy particles; one of the available technologies is the scintillator, which is generally coupled with a photomultiplier and pulse amplifier. The different particles incident on the scintillator produce electrical pulses having different shape; moreover the amplitude of these signals is related to the particle energy. The electrical pulses of the scintillator are recorded by digital acquisition/processing systems that, generally, performs a triggered acquisition consisting of a stream of pulse windows.
The aim of this study is the development of a clustering algorithm able to produce reference patterns compliant with the pattern recognition algorithm based on the matched filter technique.
The algorithm processes the data digitally acquired which contain the stream of pulse windows generated by particles having different energy and type. This paper contains a general explanation of the clustering algorithm and of the main customizations made for the scintillator signals. Many parameters can be set in the algorithm, such as the amplitude level for the negligible pulse windows or the maximum number of desired elements in the final cluster, and so on; moreover the very low probability patterns, such as the overlapping of simultaneous pulses or artefact due to noise, are removed from the final analysis.
In order to test the efficiency of the algorithm in real case, it has been applied on the data acquired during a radiation test performed at Frascati Neutron Generator for stilbene scintillator. The results show that, with this algorithm it is possible to obtain the cluster containing the neutron and the gamma shapes; furthermore, during the digital processing, for each window, the energy of the received particles and the occurrences of each different type of pulses present in the cluster are calculated.

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P2.072 Thermo-mechanical analysis of unidirectional carbon-carbon composite for thermal imaging diagnostic of a particle beam

Unidirectional carbon fiber-carbon matrix (CFC) composite tiles can be used to diagnose the main features of a particle beam such as its power, its divergence and uniformity. The diagnostic calorimeter STRIKE will be used with such aim for the negative hydrogen ion beam produced by the ITER ion source prototype SPIDER, starting operation at Consorzio RFX (Padova, Italy) in 2018. By exposing the tiles to the beam and recording their temperature with infra-red cameras the beam energy flux can be retrieved by calorimetry.
The observation of the front surface is not advisable due to the erosion of the carbon surface and to the plasma which forms in front of the tiles due to beam-gas interaction. The necessity to perform rear observation led to the choice of a very anisotropic material as CFC, which limits the broadening of the temperature profile.
The SPIDER beam heat load on each tile is constituted by 80 beamlets, carrying a total power up to 300 kW. The beamlet width depends on their divergence and the beam power density may have peaks up to 20 MW/m2, inducing large thermo-mechanical stresses. Since STRIKE is not actively cooled, beam pulse duration is limited.
In the present work a transient non-linear parametric finite element model developed in ANSYS and already validated on the basis of data collected in other test facilities is used to predict the operational limits of STRIKE, i.e. at which performance (in terms of beam power, divergence and pulse duration) SPIDER may operate when the beam is dumped onto STRIKE without the induction of

cracks in the tiles themselves. By affecting the heat transfer within the tiles in fact, such cracks would at least compromise their future diagnostic capability. The effects of different configurations for the STRIKE setup on the thermo-mechanical stresses are also discussed.

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P2.073 Study on the effects of the Signal to Noise ratio on the error in counting of the ITER Radial Neutron Camera (RNC)

RNC main purpose is to measure the uncollided 14 MeV and 2.5 MeV neutrons from deuterium-tritium (DT) and deuterium-deuterium (DD) fusion reactions through an array of detectors located in collimated lines of sight (LOS) viewing the plasma through the ITER Equatorial Port Plug #1. The measures will be used to calculate the neutron emission profile (neutron emissivity)
The environment where the RNC is be located will be very noisy, affecting the precision of the measure of the neutron and gamma particles count rates.
The paper present a study focusing on finding the maximum allowed S/N ratio, mantaining the counting error below a certain threshold (10%). Experiments will be carried out with the CAEN 5800 detector emulator, which is able to produce a particle spectra similar to the one expected in ITER.

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P2.074 Performance estimations for the ITER bolometer diagnostic

The ITER bolometer diagnostic shall provide the measurement of the total radiation emitted from the plasma, a part of the overall energy balance. Up to 550 lines-of-sight (LOS) will be installed in ITER observing the whole plasma from many different angles to enable reliable measurements and tomographic reconstructions of the spatially resolved radiation profile. The performance of the diagnostic is intimately linked to the constraints imposed by the design requirements, the sensor and data acquisition design, as well as expected noise levels.

The results presented contain the estimated signal intensities for the current design and integration of bolometer cameras in ITER, based on the simulated radiation profile for the ITER standard scenario (DT-plasma at 15 MA, 500 MW with low impurity content). The corresponding power deposited onto the absorber ranges between 0.1 μW and 0.1 W. The expected noise levels are derived mainly from requirements imposed on the signal cables, varying absorber thickness and comparisons to operating bolometer systems. Results lead to estimated values for ITER in the order of 1 μW giving a signal-to-noise ratio (SNR) which clearly indicates a need for optimizing the diagnostic.

The optimization potential is discussed. Considering the current design and installation tolerances, cameras for which collimator and pinhole parameters can be adapted to increase the light yield are identified. Most LOS with SNR<1 view the plasma edge where lower photon energies dominate. Thus, another option to increase the SNR is to investigate the potential for different absorber thicknesses as experiments demonstrated that thicker absorbers have typically higher noise levels than thinner ones. The conclusions will give recommendations for revised camera parameters to improve the light yield as well as identifying those for which an optimized absorber thickness is desirable.

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P2.075 Qualification of superconducting joint for pf6 coil

Abstract- As an important component of ITER PF6 coil, the superconducting joint is used for the mechanical and electrical connection between inner conductors and also between the coil and the outer feeder. PF6 coil plays the role in plasma current drive, position and shape control. Both the working current and the rate of working current change are high, and it works in the complex magnetic field environment in tokamak magnet system.
Before the series production manufacturing, the superconducting joint mechanical and electrical performance shall be tested and qualified. Two main samples has been designed and fabricated, one is dummy joint used for mechanical test, another is full size joint sample (FSJS) used for electrical test. This paper describes test plan and summarizes qualification though the samples. In 2017 ASIPP has started real series production manufacturing, some of manufacturing process are obtained based on the qualification.

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P2.082 Test of the Insulation Mock-up's for the ITER PF6 Coil

To qualify the insulation design to be applied on the ITER PF6 coil, two short beam-shaped mock-up's taking the form of the conductors combined with insulation in the coil have been tested in simulated Tokamak operation environment. All the results of the mechanical and electrical tests, including compressive fatigue, push-out, thermal cycling, DC and AC high-voltage withstanding, AC partial discharge, and Paschen tests, met the ITER requirement. This confirmed that the insulation design is suitable to the PF6 coil, and further the qualification of the insulation manufacture process can be started.

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P2.087 Final design and structural analysis of ITER PF5 & 6 coil assembly tools

PF5 and 6 coil assembly tool is used to transfer, support and align the PF 5 & 6 coil. The tools are comprised of PF5 lifting adapters, PF5 temporary support and align units, PF6 lifting adapters, and PF6 temporary support and align unit. PF 5 and 6 coil will be lifted from assembly hall to the tokamak pit using PF 5 and 6 lifting adapter. Then, PF5 and 6 will be temporarily placed on their temporary support and align unit. These temporary supports will be used until completion of sector assembly in tokamak pit. Then, PF5 and 6 coil will be aligned using the hydraulic jacks to their final position. The structural analysis has been performed to assess the structural integrity of the tools according to their design criteria and EN standards. PF5 and 6 lifting adapters are classified as the lifting accessory (EN 13155). PF5 and 6 temporary support and align units are classified as the steel structure (EN 1993) and the lifting table (EN 1570-1) as well, since the tools have functions of supporting and aligning. The structural stresses of the tools are lower than the allowable stress. This paper provides the design descriptions and the structural analysis results on PF 5 and 6 coil assembly tools

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P2.084 The analysis and design on transient DC over-voltage protection of ITER PF AC/DC. converter

The International Thermonuclear Experimental Reactor (ITER) Poloidal Field (PF) AC/DC. converter is composed by thyristor-based phase controlled converter modules. As the core component of ITER PF AC/DC. converter, the thyristor is very sensitive to over-voltage and could be broken down in microseconds, therefore, the transient over-voltage protection strategy is desperately essential to ensure the converter safety operation. In this paper, the transient DC over-voltage that caused by plasma disruption is studied, and a two-stage transient DC over-voltage protection strategy which combined by Metal Oxide Varistor (MOV) and external bypass is proposed based on the design principles. To ensure the reliability of the transient DC over-voltage protection strategy, a bidirectional BreakOver Diode (BOD) circuit board is designed to active external bypass when the transient DC over-voltage is positive or negative respectively, and the performance-testing platform is built to study its performance. The experiments on ITER PF AC/DC. converter test facility is also carried out, and the experiment results show that the external bypass triggered by BOD board effectively and the load current is transferred to external bypass in 2 us when BOD suffers an over-voltage. The effectiveness of the proposed transient DC over-voltage protection strategy is verified.

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P2.090 Design of the inverter based on IEGT of acceleration grid power supply for CFETR N-NBI prototype

China has started the research and development of the negative-ion-based neutral beam injection (N-NBI) prototype for Chinese Fusion Engineering Testing Reactor (CFETR). The prototype needs an acceleration grid power supply (AGPS) rated at 200kV/25A/3600s. A single stage inverter-type high voltage power supply is applied as AGPS of the prototype. The AGPS consists of phase-controlled rectifier, inverter, step-up transformer and high voltage uncontrolled rectifier. The inverter is the core component of this AGPS and determines the performance of the AGPS directly.
Injection enhanced gate transistor (IEGT) is a voltage-controlled device and has high switch frequency, low loss, simple gate drive and wide safe operating area (SOA). The inverter of the AGPS is designed with press pack IEGT in three phase three level (TPTL) neutral point clamped (NPC) topology, and regulated by duty cycle modulation with an inverter frequency of 150 Hz. The low inductance structure of the main circuit of the inverter can guarantee the maximum surge voltage of the IEGTs at a safety level. In order to obtain the low inductance structure, the electromechanical structure and the water cooling system of the inverter are carefully designed by structural simulations, electromagnetic simulations and thermal simulations. What's more, the drivers of the IEGTs adopt numerical controlled active clamp technology and segmented drive technology to switch off all IEGTs less than 50μs and ensure the safety of the AGPS when breakdown or beam-off occurs between the acceleration grids.
The detailed design of the inverter for CFETR N-NBI prototype AGPS is described in this paper. A water-cooled 3-level phase leg assembly has been constructed and tested. It has proven that the inverter designed in this paper has good performance and high reliability.

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P2.086 A novel series switch module in high-voltage applications

Due to safety and reliability concerns, high voltage fast switch is required for Tokamak power system. It can cut off the high voltage power supply within a little time usually in microsecond, when fault occurs, system tests and load switches. IGBT is an ideal choice as the basic component of the switch

to satisfy the fast response requirement. Obviously, connecting multiple IGBTs in series directly is an essential approach in high-voltage applications. But voltage balance between IGBTs connected in series is a very complex and difficult problem, and there is no satisfactory method so far. This paper presents a novel switch circuit module, which can operate easily as a modular-connecting high-voltage switch. Based on the improved MMC topology, its switch time and voltage adjustment ability are advanced and more. When module turns off, it inhibits overvoltage by a high capacity capacitor after voltage rises to rated voltage. In order to balance the voltage when switch is in conducting state, two IGBTs in overvoltage-module turn on. In the meantime, this operated mode increases current carrying capacity. The test prototype is a 10000V/100 A switch with 4 modules in series. The simulation and experimental results show that this switch can effectively improve the dynamic-voltage balance and shorten switch time. All in all, the work presented in this paper has a significant reference value for practical engineering applications.

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P2.091 Design of acceleration grid power supply for CFETR negative-ion-based neutral beam injection prototype

In order to form an independent capacity of design and engineering construction for negative-ion-based neutral beam injection (N-NBI) system, and lay the foundation for the construction of Chinese Fusion Engineering Testing Reactor (CFETR). China has started the research and development of the N-NBI prototype for CFETR. The prototype is designed to accelerate hydrogen negative ions up to −200keV with a beam current as high as 20A for 3600s, and need an acceleration grid power supply (AGPS) rated at 200kV/25A/3600s.
A single stage inverter-type high voltage power supply, which is powered by 10kV ac power grid, is applied as AGPS of the N-NBI prototype. The AGPS includes a controlled rectification system, which is composed of a step-down transformer system and a 24 pulse rectifier based on thyristor. The controlled rectification system converts the ac 10kV of the power grid into dc 5000V to feed a dual-dc-link system. This dual-dc-link feeds a three phase three level (TPTL) neutral point clamped (NPC) dc/ac inverter which based on injection enhanced gate transistor (IEGT) and regulated by duty cycle modulation with an inverter frequency of 150 Hz. At the output of the inverter, a set-up transformer with turn ratio of 1:23.6 is used to increase the voltage level and to provide the required insulation between high voltage side and low voltage side. A high voltage uncontrolled rectifier based on avalanche diode transforms the ac high voltage of the transformer output into dc high voltage. And a high voltage R-C dc-filter smooths the dc output voltage.
This paper describes the detailed design of the AGPS for CFETR N-NBI prototype, focusing on the inverter, the step-up transformer and the high voltage diode rectifier. And the reliability of the design is verified by circuit simulations and electromagnetic simulations.

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P2.092 FPGA-based interlock system for the chopper of the Linear IFMIF prototype accelerator injector

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The Linear IFMIF (International Fusion Materials Irradiation Facility) Prototype Accelerator (LIPAc) injector consists of a 140 mA proton/deuteron source, its associated low energy beam transport line (LEBT) as well as ancillaries such as water cooling skid, vacuum groups, High Voltage Power Supplies (HVPS), etc . A specific element, the beam “Chopper”, was included in the LEBT to generate short (~ 100µs) and sharp edged beam pulses (~10µs) and allow the use of interceptive diagnostics

in the high energy part of the LIPAc during commissioning phases of the RFQ (5MeV) and the SRF Linac (9MeV). The chopper was designed to operate in pulsed mode with very sharp rise and fall times, meaning the chopper will be used to "cut" the long rise time of the source as well as the fall time of the beam pulse.
The chopper thermal screen has not been designed to withstand very high beam power (i.e. beam length and duty cycle need to be monitored); in addition, the chopper HVPS needs to be monitored in real time to detect a possible trip and extract the beam before downstream devices are damaged. For these applications, standard PLC based interlocks are too slow and faster solutions are envisaged.
The proposed solution for the required interlock system is based on COTS technology based on XILINX FPGAs using RIO (Reconfigurable Input/Output) technology from National Instruments (CompactRIO platform), that provides reconfigurable FPGA-based embedded system for control and data acquisition applications. The paper describes the implementation of the interlock system, the response times of the proposed architecture and the performance of the system. Also, the CompactRIO contains a microcontroller running a real-time Linux operating system and uses LabVIEW Channel Access Server and Client to interface with EPICS in order to monitor and control the interlock system.

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P2.093 Models for the power grid load at JET

The JET tokamak is connected to the United Kingdom 400kV National Grid by three Super Grid Transformers (SGTs) through a 36kV power network. The 36kV system supplies power to the toroidal (TF) and poloidal (PF) field coils, heating systems and their auxiliaries. The voltage drop on the system is limited. Dropping below 30kV will first trip the heating systems followed by a total loss of power below 29kV requiring 8 hours of recovery time.
The JET power supply system can handle large loads with full neutral beam power (up to 35MW) and high toroidal fields (up to 4T) in its standard configuration with three SGTs connected and the toroidal field supplied by both a flywheel generator and static units. In 2016 a flashover in one of the SGTs, reduced the operational space to two super grid transformers and required precise power load profile predictions and pre-pulse checks to stay above the power supply trip levels.
This contribution describes the theoretical and statistical models of measurements that were used in the power load predictor. The study was based on 86 dry runs (magnetic test pulses) in 2012-13 (#83499-85001) and 4302 pulses (#86495-90796) in the C33-C36 JET campaigns. Power load models were obtained using statistical model selection methods both on the total apparent and reactive power as well as on each power supply individually. The power load predictor was extensively used in 2016 until the repair of the third super grid transformer. It enabled the optimisation of plasma discharges and full exploitation of the limited operational space available at the time.

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P2.096 Receiving Inspection and Tests (RIT) of ITER Poloidal Field #4 Cryostat Feedtrough in MIFI.

The ITER Magnets are in the procurement and assembly phases.
During these phases, critical components need to be tested and assembly procedures need to be developed and qualified on mock-ups.
To this end, ITER Organization (IO) and CEA have built up a support structure with space, expertise and equipment: MIFI – Magnet Infrastructure Facilities for ITER.
In this framework, IO and CEA perform Receiving Inspection and Tests (RIT) on components delivered to IO by the Domestic Agencies (DA).
These tests are meant to ensure the good quality of components and occur between the Manufacturing Inspection Tests that are performed before delivery to ITER and the Site Acceptance Tests that are performed before installation in the tokamak pit.

The goal is to verify that nothing critical occurred during the shipment and to check original data from the DA.
The paper describes the RIT sequence of the Poloidal Field #4 Cryostat Feedthrough (PF4 CFT). A detailed Manufacture Inspection Plan (MIP) has been especially written to check critical parameters. A particular focus is made on three topics: metrology, leak detection and high voltage (HV) tests.
First, the dimension and geometric tolerances are checked with a laser tracker and compared to the requirements of the feeder project.
Then, leak tests are performed. The vacuum barrier (inside the PF4) must be leak tight to guarantee different levels of vacuum, internal piping as well as vacuum duct welds must be verified.
Finally, HV tests on busbars will verify the specification for the resistance, the continuity and the insulation.
The RIT results and the improvements proposed on the Manufacturing tests will be presented.

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

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P2.097 Prediction of mechanical behavior of the KSTAR CS magnet based on PF coil currents

The KSTAR has seven pairs of ring-shaped poloidal field (PF) superconducting coils with rectangular cross section. The central solenoid (CS) is a vertical stack of four pairs of PF coils compressed axially by preloading structures. The axial compression (remaining preload) on the CS coils is monitored by a strain measurement, one of the important monitoring parameters for safe operation of KSTAR. The preload variation during plasma discharge can be predicted based on the PF coils and plasma currents without complicated structural analysis. For the prediction, the relations between stiffness of CS coil and structure, preloading, equivalent electromagnetic (EM) force, and displacement have been established using elasticity, experimental data and finite element (FE) analysis. The EM force can be quickly calculated using the superposition of EM analyses results for unit current of each PF coil and plasma. The relations were verified through the application to the case of a long pulse plasma discharge. This study will be helpful to provide the optimal combination of PF coil current limits for future KSTAR operations.

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P2.098 Protection system for Wendelstein 7-X superconducting magnets

The Wendelstein 7-X stellarator (W7-X), one of the largest stellarator fusion experiments, will start the third plasma operation campaign mid of 2018 at the Max Planck Institute for Plasma Physics in Greifswald, Germany. The main objective of the experiment is to prove the reactor relevance of the stellarator design.
The W7-X experiment has a superconducting magnet system with 50 non-planar and 20 planar coils grouped in five equal modules and electrically connected in seven circuits with 10 coils each. Seven power supplies provide individual coil currents in the seven circuits. A quench detection system checks permanently the superconducting system regarding the onset of a quench. In case of a quench or a severe failure, the magnet protection system will be activated and leads the current into dump resistors. The magnet protection system consists of a set of switches and breakers and a dump resistor, made of pure nickel. The system was original designed for currents up to 20 kA and voltages up to 8 kV with a total energy of about 1 GJ. Based on new requirements and operation experiences, updates and improvements of several components were implemented in the last 12 month.
The paper will present the design and function of the magnet protection system and describes the

motivation, the studies as well as the implementation and tests of updated components and procedures in detail.

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P2.099 ITER magnet coil power conversion building construction and AC/DC converter installation

ITER (IO) magnet coil power supply system is the world largest AC/DC conversion system which is jointly contributed by China, Korea including PF, CS, VS, TF and CC AC/DC converters with total capacity 2853 MVA and the associated Fast Discharge Unit, Switch Network Unit and DC Busbar from Russia Federation. All ITER coil power supply AC/DC converters will be installed in the magnet power conversion buildings (MPCB) constructed by European Union.

MPCB includes two framed structured buildings No.32&33 with seismic classification of SC2. Each building consists of the volume of 150m long, 31m wide and 7.5 m high hosting two groups AC/DC converters along the building on north and south. The construction started from October 2013 which is approaching Ready for Equipment (RFE) in august 2018.

As a part of the converter design and manufacture, the installation of AC/DC converters was taken into account during the life cycle of AC/DC converter development particularly on design review and manufacture stages which goes in parallel with MPCB design and construction. 32 AC/DC converter units for ITER project Phase-I will be installed and integrated on IO site from 2018 after the building RFE till 2021. Technically the interface between building construction and all sorts of DA converter components are managed by IO. As the early installation, 5 PF transformers from China had been positioned on transformer foundations as per the early access of MPCB in 2017.

This paper describes the ITER magnet power conversion buildings, briefs the construction progress and RFE conditions. The converter integration and installation as well are detailed in terms of layout configuration, installation sequence, schedule integration, site management and coordination.

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P2.101 Busbar Monitoring System of the ITER DC Busbars

In ITER, DC busbars will be used to connect the AC/DC converters to the superconducting coils of the magnet system and will run from buildings B32, B33 through building B74 to TOKAMAK building B11; the total length will exceed 5 km. The busbars will be interconnected by flexible links for thermal expansion compensation.

The busbars used in the TF, PF/CS and CC coil power supply systems are rated for the continuous current up to 68 kA, 55 kA and 10 kA respectively. The busbars will be cooled by deionized water through inlet and outlet cooling water collectors (CWC). The water flow rate and temperature will be measured by sensors installed in outlet CWCs. Reduction or loss of cooling water flow might cause excessive heating of the busbar conductors and, in extreme case, damage their insulation. To prevent this, the Busbar Monitoring System (BBMS) was developed at the Efremov Institute for continuous monitoring of thermal conditions of the busbars during operation with a polling frequency of 1 Hz.

BBMS is composed of two independent subsystems, namely, one subsystem collects data from the CWCs, the second one measures temperature of the flexible links surface by fiber optic temperature sensors (FOTS). The CWC data acquisition subsystem is related to the Interlock system, and in case the water cooling system fails it automatically generates event for controllable interruption of the plasma pulse. The FOTS data acquisition subsystem is used for back-up diagnostics of the busbar

cooling conditions and for estimation of condition of the contacts between the busbar sections and flexible links.

This paper presents the architecture of BBMS after a short description of the ITER DC busbar system. Special attention will be paid to the detailed description of the BBMS components and results of their testing, as well as to the software description.

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P2.103 Synergistic effects of nitrogen seeding impurities and transient heat loads on tungsten

The synergistic effects of transient heat loads in conjunction with stationary plasma were investigated. All tests were executed in the linear plasma device PSI-2 at a base temperature of around 800 ℃, which was achieved by the plasma exposure and an ohmic heater attached to the sample holder. Moreover, to simulate the ELM-like transient thermal events, a Nd:YAG laser with a wavelength of 1064 nm was used. The impact of up to 1000 transient thermal events with an absorbed power density of 0.38 GW/m², a pulse duration of 1 ms, and a repetition frequency of 0.5 Hz was analysed. A mixed D/He/N plasma with a flux of around 8 × 10^21 m^−2s^−1 was used to realize the stationary plasma background.
The obtained results of the D/He/N seeding impurity experiments are compared with D/He results and indicate that N, during low pulse number tests (≤ 1000 pulses), seems to have no significant influence on the thermal shock performance of tungsten in terms of damage formation such as roughening due to plastic deformation or cracking. However, particle induced nano-structures on the laser/plasma and, less pronounced, on the only plasma exposed surface show that N leads to the formation of lamella like structures. These structures could originate from glide planes, which become visible on the surface. Furthermore, the particle induced bubble formation below the exposed surface disappears or is significantly reduced if N is present in the plasma. A possible explanation for both effects could be the higher erosion rate compared to the H/He exposure. Especially the reduction or even absence of bubbles below the exposed surface could have a significant influence on the lifetime of the plasma facing material since a zone with a high bubble density below the surface could lead to a reduced thermal conductivity and therefore to overheating/melting.

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P2.106 Heat load testing of 3D printed PFC mockups to evaluate thermo-hydraulic enhancement

ITER and DEMO plasma facing components (PFCs) should remove the extreme heat flux up to 10 MW/m^2 and the various type of PFCs have been developed for enhancing the heat transfer performance such as hypervapotron and twisted tape insertion. For the limitation of complexity to mechanical machining, three-dimensional (3D) metal printing technology by direct energy deposition is used to fabricate the multi-layer cooling devices for the development of PFCs research in Korea, also for the optimization of the thermo-hydraulic performance with the water cooling in a Korean heat load test facility by using an electron beam (KoHLT-EB). Two-type's cooling structure were fabricated with aluminum-bronze metal powder by using 3D printing method, one type has the flat cooling channel with an elliptic shape, and the other type has the swirl tape with a hollow shape. Preliminary thermo-hydraulic analysis was performed to confirm the effects of cooling geometry inside with a conventional CFD code, ANSYS-CFX. KoHLT-EB facility was used to evaluate the enhancement of the cooling performance, and the heat load was applied from 1 to 10 MW/m^2 to evaluate the thermos-hydraulic performance for each enhanced channel device. The present research results will contribute to the development of a Korean fusion reactor and DEMO program.

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P2.107 Cleaning of the surface of beryllium armor tiles and beryllium products from beryllium-content particles

The total area of the first wall (FW) of the International Thermonuclear Experimental Reactor (ITER) is 650 m2, 40% of which is the responsibility of the RF DA. The NIIEFA should manufacture and test 179 first wall panels (FWP), which requires 7000 high-heat-flux units and 305 000 beryllium tiles. As is known, beryllium is a toxic material, therefore stringent requirements are imposed on the permissible content of beryllium on the surface of products.
This article presents the comprehensive work performed to investigate the concentration of impurities on the surface of beryllium products. Various methods for cleaning of beryllium armor tiles and beryllium products from beryllium-content dust after thermal tests have been tried out (ultrasonic cleaning, cleaning with surface-active agents, mechanical brushing) and the impurity concentrations on the surface have been defined by the colorimetric method. The phase and element composition of the surface contamination has been studied. Experiments have been carried out to define the dependence of the impurity concentration on the storage time of beryllium armor tiles and beryllium products.
It has been found that integrated cleaning in an ultrasound bath with multi-stage rinsing ensures the best results. Optimal regimes for ultrasonic cleaning of the beryllium tiles have been developed, which provide, first, the minimum content of removable beryllium-content impurities on the surface and, second, reproducibility of the cleaning results under the conditions of serial production of beryllium products. Further works on cleaning and measuring of the surface contamination of beryllium products after thermal testing should be carried out, as their surface is complicated in geometry and large in area.

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P2.109 EM analysis and thermal-mechanical dimensioning of First Plasma Protection Components

The First Plasma Protection Components (FPPC) are a set of structures temporarily installed inside the vacuum vessel specifically for first plasma operation. In addition to aiding plasma generation, these components protect the unshielded vacuum vessel and ancillary systems from the plasma. Upon completion of first plasma operation, these components will be removed.
A significant design and analysis activity had been performed during the conceptual design phase.
For the final design phase, a revised design is being developed, from a combination of feedback from the conceptual design review, changes to interfacing requirements with other systems, and further investigation into detailed aspects of the design.
The feasibility of the final design concept of the four FPPC components (Temporary Limiter, Divertor Replacement Structure, ECRH mirror and dump) needs to be verified through an exhaustive set of thermal, electromagnetic (EM) and stress analyses before being presented to the Final Design Review.
The analysis includes evaluation of EM loads during reference plasma disruption events and of the temperature evolution due to the surface heat flux coming from the plasma.
The EM and thermal loads are used as input for the stress analysis of the FPPC, together with bolt preloads and seismic loads for the evaluation of static strength of the parts in accordance with the specified ITER design criteria (called SDC-IC).
The structural finite element analyses use static non-linear approaches, assuming the elastic or elastic-plastic behaviour of the materials. Further analyses make extensive use of the structural sub-modelling technique for detailed analysis of bolts connections.
This paper outlines the engineering aspects of the FPPC and focuses on the feasibility of the present design by structural assessment.
The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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P2.112 Assessment of mechanical properties of tungsten wires for the fiber-reinforced plasma facing components

Recent efforts dedicated to the mitigation of tungsten (W) brittleness have demonstrated that tungsten fiber-reinforced tungsten composites show toughness even at room temperature. This is caused by extrinsic mechanisms induced by the incorporated tungsten wire used as reinforcing element. High temperature operation and manufacturing of the fiber-reinforced composites might result in a change of the mechanical properties of the wires and thus to a change in the overall composite properties.
To address the issues related to the fiber properties, we investigate mechanical and microstructural properties of pure and potassium-doped tungsten wires, being heat treated in the temperature range between 900-2300℃. The mechanical tests are performed starting from room temperature up to 600℃. The microstructure of the wires in as-received, as-annealed and as-deformed conditions is investigated by a number of techniques including scanning and transmission electron microscopy employing focus ion beam for the sample preparation.
Based on the light optical microscopic analysis, the engineering deformation curves are converted into actual stress - strain dataset, accounting for the local necking. The analysis demonstrates that local strain in the necking region can reach up to 50% and the total elongation monotonically increases with temperature, while the ultimate tensile strength goes down. High temperature annealing above 1600℃ makes the wire completely brittle, while K-doping shifts the threshold annealing temperature up to 2100℃. Mechanical tests performed at elevated temperatures show that the ultimate tensile strength of the wires exhibits drastic reduction for the pure W wires annealed above 1300℃. K-doped wires sustain considerable tensile strength up to at least 1900℃. K-doping clearly demonstrated the improved resistance against high temperature annealing otherwise causing sever degradation of mechanical properties in pure W wire.

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P2.113 Heat transfer enhancement in MHD free-surface flow by controlling the electromagnetic force with fin structure

Partly insulated fin structure has been proposed to mitigate the temperature stratification in the flowing-type liquid metal divertor. This structure consists of partly insulated fins which are infused in the flowing liquid metal. Our previous study observed the generation of a wavy flow by a checkerboard like arrangement of insulated parts experimentally and numerically. Moreover, magnitude of the wavy flow increased with a rise in the magnetic field. However, the experiment was conducted using a closed channel due to the magnetic field direction of the experimental apparatus. Therefore, in this study, three-dimensional magnetohydrodynamics (MHD) flow simulation was conducted using an analytical model which has the free-surface to clarify the flow field in this concept.
The analytical model simulated two partly insulated fins which are infused in the liquid metal free-surface flow. Height of fins and liquid metal were 20.0 mm and 30.0 mm respectively. Streamwise length of the model was 120 mm and the insulated parts were switched every 30 mm alternately. Governing equations were Navier-Stokes equation and Poisson equation of the electric potential. Applied transverse magnetic field was up to 0.2 T due to the nonlinearity of the electromagnetic force term in the governing equation. Free-slip condition was applied on the free-surface and deformation of it was not considered.
Results showed that a ratio of the vertical component of velocity to streamwise component was 30% at the maximum. In addition, the wavy flow generated by the partly insulation reached near the free-surface region. This indicates that the structure possibly enhances the heat transfer near the free-surface. Evaluation of free-surface deformation using Arbitrary Lagrangian-Eulerian method and results of heat transfer simulation are also planned to be presented at the conference.

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P2.116 Inverse identification of Tungsten static recrystallization kinetics

For the ITER divertor, plasma facing components are made with tungsten as armour material, bonded on a coper alloy tube as heat sink structural material and cooled by water. Such components withstand high heat flux up to 20 MW/m2 and consequently satisfy ITER requirements. However, due to high heat flux, the loaded surface can reach extreme temperature values from 2000℃ and strong temperature gradients are generated on a thickness of 6 mm. 2000℃ is large enough to alter the tungsten microstructure by recrystallization causing mechanical properties losses and then damages such as macro cracks in the material. Understanding of recrystallization phenomenon is essential to apprehend damage process of tungsten armored components and to optimize their use in tokamak environment. Previous, numerical studies proved that recrystallization phenomenon play an important role in the damage process of the tungsten armored component. Up to now, uniform tungsten recrystallized layer thickness was assumed in numerical simulations to predict plastic strain increments and then perform life time calculations.
In this paper, a way of modelling tungsten recrystallization is proposed. To reach this goal, experiments are performed to study recrystallization kinetics of tungsten up to 1800℃ thus completing experimental data available in literature limited at 1350℃ [1]. Each obtained recrystallization kinetic is fitted to Johnson-Mehl-Avrami-Kolmogorov (JMAK) model. This phenomenological model is able to describe recrystallization kinetics as an exponential function useable in finite elements code. In this way, based on isothermal recrystallization kinetics gathered by A.Alfonso et al [1] and our experiments, numerical simulations coupled with JMAK model are achieved to model tungsten recrystallization. Using these experiments data and dedicated simulations coupled with JMAK model, consistent recrystallization gradient are estimated compare to that obtained experimentally during high heat flux campaigns.

[1] "Alfonso Lopez et al, (2015). Thermal stability of warm-rolled tungsten. DTU Mechanical Engineering".

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P2.117 Coupled fluid-structure thermofluid-dynamic analysis of the DEMO divertor cassette body equipped with a liner

Within the framework of the Work Package DIV 1 - "Divertor Cassette Design and Integration" of the EUROfusion action, a research campaign has been jointly carried out by University of Palermo and ENEA to investigate the thermal-hydraulic performances of the DEMO divertor cassette cooling system. The research activity has been focussed onto the most recent design of the Cassette Body (CB) cooling circuit equipped with a Liner, whose main function is to protect the underlying vacuum pump hole from the radiation arising from the plasma. The research campaign has been carried out following a theoretical-computational approach based on the Finite Volume Method and adopting the commercial Computational Fluid-Dynamic code ANSYS-CFX.
The CB thermal-hydraulic performances have been assessed in terms of coolant and structure temperature, coolant overall pressure drop, flow velocity distribution and mass flow rate fed to the Liner cooling circuit, mainly in order to check coolant aptitude to provide a uniform and effective cooling to both CB and Liner structures.
The outcomes of the study have shown some major criticalities, mainly in terms of water coolant vaporization as well as of non-symmetric coolant distribution between the two Liner inlets. As a consequence, the following potential solutions have been successfully explored in order to allow the CB to safely operate complying with its design constraints:
• revising the CB design layout in order to increase the coolant mass flow rate fed to Liner;
• increasing coolant inlet pressure to rise water saturation temperature and, hence, its margin against vaporization;

• increasing coolant mass flow rate to reduce its overall thermal rise.
The main results and the achieved optimized model are herewith described and critically discussed.

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P2.120 The adhesion of tungsten dust on rough tungsten surfaces

Adhesion plays a pivotal role in numerous aspects of tokamak-generated dust such as in-situ removal techniques, post-mortem collection activities, resuspension during loss-of-vacuum accidents and in-plasma remobilization. Due to insurmountable difficulties in the theoretical treatment of the interaction between technical (rough, polycrystalline, adsorbate covered) surfaces, adhesive or pull-off force measurements for reactor relevant materials are imperative.
Recent measurements of the adhesion of spherical micron W dust (R_d⊠5-15μm) on planar W surfaces (R_q⊠10-100nm) revealed the importance of surface morphology. Due to nano-roughness: (i) Adhesion is not dominated by metallic bonding interactions but by London dispersion forces, implying that the Van der Waals expression can provide a zero-order estimate of the pull-off-force. (ii) Adhesion is not deterministic but acquires a statistical nature and is better described by cumulative distribution functions. In particular, the pull-off force behaves as a log-normally distributed random variable.
However, in situ tokamak roughness is characterized by much higher root-mean-square values. Surface roughness of the order of the contact radius (R_q~1μm) should lead to a respectable decrease of the mean value of the measured pull-off force and increase of its relative standard deviation. To quantify adhesion for tokamak-relevant surfaces, we have performed systematic pull-off force measurements for W dust deposited in a controlled manner on rough planar W surfaces (R_q~1μm). Monodisperse spherical W dust (R_d=9,15μm) was adhered to rough W samples through impacts below the sticking threshold. The samples were then inserted in an electrical mobilization setup that allows the measurement of the pull-off force by exerting an electrostatic force of well-known magnitude. The measurements revealed a ~50% reduction in the mean value and a ~100% increase in the relative standard deviation of the pull-off force for rough surfaces (R_q⊠1μm) compared to smooth surfaces (R_q⊠10nm). Despite these differences, the pull-off force still follows the log-normal distribution.

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P2.126 ITER-grade tungsten limiters damage under high heat flux in the T-10 tokamak

Tungsten is foreseen as plasma facing material for ITER. The tungsten testing under high heat loads is very important for ITER operation prediction. In a real tokamak conditions combination of the heat and particle fluxes could enhance tungsten destruction and erosion. Some mechanisms of plasma-surface interaction can lead to a concentration of heat flux onto the small zones of the divertor and limiters and cause a damage or melting of tungsten. One of such mechanisms could be nonambipolar flow toward the tungsten surface due to the arcs and sparks. This report presents the results of the tungsten limiters analysis in the T-10 tokamak after 2015-2017 experimental campaigns. The limiters were made from the ITER-grade WMP "POLEMA" tungsten.
Significant cracking of the tungsten limiter surface was observed. Heat load up to 2 MW /m2 lead to the micro-cracks (1-2 μm wide) development at the grain borders accompanied by the destruction of the surface. On the inner board of the torus heat flux to the edge of the limiter ring can exceed 5 MW m-2 during the ECR heating. In such discharges, the tungsten plates at the inboard of the vessel were heated to temperature exceeded 20000C. Local thermal load of more than 40 MW/m2

leads to tungsten surface melting. Intensive sparking and arcing, deep cracks and edge melting were observed on the tungsten tiles after the experimental campaigns. In these zones macro cracks developed with a width up to 100 μm and lengths of 5-50 mm. It is supposing that the nonambipolar flow due to the arcs and sparks leads to the observed overheating, cracking and melting of the tungsten surface.
In the T-10 tokamak edge plasma conditions strong destruction of the ITER-grade tungsten limiters was observed during the experimental campaigns.

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p2.124 Maximizing JET divertor mechanical performance for the upcoming Deuterium-Tritium experiments

The next Deuterium-Tritium campaign will push JET ITER-like wall to high divertor power and energy levels. During the 2016 campaign, the Strike Point (SP) sweeping technique enabled us to run relevant H-mode scenarios without exceeding the temperature limits imposed by JET Operation Instructions (JOIs). In the subsequent shutdown, six outer divertor tungsten-coated 2D carbon fibre composite (CFC) tiles were found to have radial cracks between the fibre planes.
In this paper we describe the results of a 3D thermo-mechanical analysis aimed at understanding the origin of the cracks. A sensitivity assessment has been carried out for modelling the temperature time-evolution on a realistic tile 3D model during high power shots, both with fixed and sweeping SPs. Time and space-varying heat flux density loads on the divertor surface have been calculated using different techniques for reconstructing the power engineering footprint. The results have been benchmarked against the temperature measurements by high resolution infrared (IR) diagnostic systems.
The study has confirmed the source of the cracks and their localization in the upper part of the tile, giving CFC ply normal direction stress values higher than the Ultimate Tensile Stress allowable for this composite material.
Other parameters have been found playing an important role on the stress values arising in the tile. An intrinsic structural criterion has been found that relates, at the same time, the input power, the radial location of the SP and the amplitude of the sweeping tile surface. Using this criterion as threshold, the optimal combination of these parameters has been calculated which maximises the performance of divertor tiles without exceeding the limits on input energy and maximum surface temperature imposed by the JOIs.

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P2.125 Investigation of deuterium retention in tungsten exposed to high-flux steady-state helicon plasmas

High Magnetic field Helicon experiment (HMHX) is a linear helicon wave plasma (HWP) source with high axial magnetic field (B0<6300 G), which address fuel retention in first wall materials. High flux Ar/D2 plasmas are produced using an inner half helical antenna with RF power source operating at a frequency of 13.56 MHz at power levels up to 5 kW. Langmuir probe, OES and Hiden EQP (Electrostatic QuadruPole Plasma Mass Spectrometer) analyzer are used for the measurement of EEPFs, IEDF of deuterium ion D+ and other plasma parameters (including electron density ne, electron temperature Te, plasma potential Vp). The ion flux and energy of D+ are controlled by tuning the flow-rate ratio of D2 and Ar. Tungsten samples are exposed to 30 mins plasma pulses in an axial magnetic field of 1300 G with different substrate temperature and negative bias voltage. Particle balance analysis is used to estimate the real time evolution of the deuterium in-vessel retention. Deuterium retention in tungsten samples are investigated later using thermal desorption spectroscopy. Results shows that deuterium retention in HWP mode is much higher than that in ICP (Inductively Coupled

Plasma) mode due to higher ion flux and energy. Also, a strong reduction of retention with higher surface temperature and lower negative bias is found and an approximate model is used to simulate diffusion and trapping of hydrogen in tungsten.

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P2.129 Structured Cooling Channels for Intensively Heated Blanket Components

The cooling performance of surface structuring for enhancing heat transfer in cooling channels of helium-gas cooled First Wall applications and their prospects of success in efficiency and effectiveness were investigated for several thermal-hydraulic conditions and structure designs in the last years. Cooling channel structured by upstream and downstream directed, truncated 60° V-shaped ribs, by semi-detached 60° V-shaped ribs and by spherical dimples were investigated. The structure design was developed under consideration of thermal, mechanical and fabrication aspects. The heat flux density was in the range from 0.5 MW/m2 to 1.0 MW/m2 as expected to occur on the first wall and the helium mass flow rate was varied from 0.023 to 0.070 kg/s. The results can be summarized as follows:

- Based on numerical results, engineering correlations for heat transfer coefficients were evolved and can be used in thermal-mechanical analysis and design of cooling channels.
- An efficiency criterion for evaluating the cooling performance of structured cooling channels shows that the pumping power is significantly reduced for the structured cooling channels to obtain an equivalent cooling performance when compared to smooth cooling channels.
- Lowest friction factors occur for the dimpled channel walls, but V-shaped ribs perform better than spherical dimples due to the enormous heat transfer enhancement.
- Possible fabrication concepts for the surface structures were presented.

It is concluded that surface structuring is a viable approach to make helium cooling of thermally highly loaded plasma facing components feasible, and can also improve the thermal performance of blanket internal cooling.

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P2.130 Armoring the Wendelstein 7-X divertor observation immersion tubes against NBI orbit losses

The first neutral beam injector (NBI) experiments of the Wendelstein 7-X stellarator will start in summer 2018. The modelling of the fast ion production and slowing down processes [1,2] predicts losses of the NBI fast ions to the first wall on the order of 15%. One location receiving a high load (possibly peaking at several MW/m2) is the immersion tube for optical and infrared monitoring of the divertor targets. In the present design, the cooling of each tube is only meant to remove the heating by plasma radiation as well as the heating from the cameras inside the tube. The stainless steel face of the tube has three vacuum windows, which are sensitive to temperature gradients and overheating. To protect the windows from damage caused by the fast ions, different heat load mitigation techniques were investigated. Given the available time and resources until the first NBI experiments, a

protective collar mounted at the front of the immersion tubes is regarded the most realistic solution. This contribution describes the fast ion modelling of the loads, the new design, thermal modelling of the design, and finally experimental experience with the protective collar. The fast ion heat loads have been assessed with the ASCOT code [3], the thermal loads with field line diffusion calculations [4] and the thermal response with the COMSOL multiphysics FEM code.

[1] Drevlak et al 2014 Nucl. Fusion 54 073002
[2] Äkäslompolo et al Nucl. Fusion (Under review, Proceedings of the 15th IAEA TM on energetic particles)
[3] Hirvijoki et al, CPC 185 (2014) 1310–1321
[4] Bozhenkov et al FusEngDes 88 (2013) 2997–3006

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P2.022 Initial port integration concept for EC and NB systems in EU DEMO tokamak

The integration of heating current drive (HCD) systems in the EU DEMO tokamak must address a number of issues, namely space constraints in the tokamak building, remote handling requirements, breeding blanket penetration, neutron and photon radiation shielding, compliance of penetrations of the primary vacuum with safety and vacuum criteria, and a large number of loading conditions, in particular heat, electromagnetic (EM), and pressure loads in normal and off-normal conditions. A number of pre-conceptual design options of the vacuum vessel (VV) port and the port-plug are under assessment and need to be verified against all requirements and related criteria.

The identification of the functional (or physics) requirements of the HCD systems remains an ongoing process during the pre-conceptual design phase, hence some initial assumptions had to be made as a basis for development of the design of the vacuum vessel ports and the HCD port plugs.

The paper will provide an overview of present margins in the functional/physics requirements and the rationale behind the assumptions made in order to allow development of the pre-conceptual design options. Furthermore it will introduce the initial design concepts of the Electron Cyclotron (EC) Launchers and the Neutral Beam (NB) Injectors integrated in the equatorial ports. The NB duct design in DEMO and related issues such as transmission and re-ionization losses will be also addressed.

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

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P2.135 Study on the Endoscopic Inspection of ITER Thermal Shield Cooling Pipes

A novel endoscope has been developed for the inspection of long and complex-shaped cooling pipe of ITER Thermal Shield (TS). The mechanical design has been improved and its endoscopic images are clarified for various pipe surface conditions. Main break-through is to reduce the cable friction against metal pipe inner surface as well as to maintain the elastic rigidity of the cable for insertion into long cooling pipe. Spacers are attached on both the camera head and the cable to reduce the friction. Plate spring tube is used for the cable for the elastic rigidity. The conceived endoscope is tested to the TS pipe routing mock-up, which is longest and most complex among the TS pipe routings. Special device is also presented for making the endoscope insertion be easier. The insertion

test result is compared with that of a conventional endoscope. The difference between the images from the endoscope and the real pipe surfaces are also discussed for various kinds of samples.

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P2.138 Design of the K-DEMO in-vessel blanket arrangement blanket sector maintenance details and upper lever RM enclosure

A number of blanket arrangements and maintenance options are being investigated within various fusion DEMO studies. The defined segmentation, general arrangement and maintenance approach for the vacuum vessel blankets has a major impact on the overall device configuration. The K-DEMO blanket arrangement is centered on an approach to minimize the number of blanket segments that are accessed through vertical maintenance ports. Minimizing the segment number reduces the number of piping services that interface with it and exit the vacuum vessel. The design definition of interfacing components was developed to enhance this plan. Concept design details of the prescribed K-DEMO machine configuration have evolved in sufficient detail to allow the merits (and shortcomings) of this design approach to be evaluated with respect to other blanket segmentation and maintenance options. Coupled with the defined in-vessel blanket arrangement and maintenance approach a conceptual design of an intermediary maintenance enclosure above the machine core has been developed to interface, handle and move blanket sectors from the device core through vertical ports without using individual cask enclosures located at each vertical port opening. A structural assessment has also been developed to determine how well the planned blanket support system design fairs with respect to a representative disruption load test case. The results of this activity along with an overview of the latest K-DEMO device general arrangement will be presented.

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P2.139 Preliminary configuration of the torus vacuum pumping system installed in the DEMO lower port

In the European DEMO program, the design development of the demonstration power plant is currently in its pre-conceptional phase. This work includes also the design development of the vacuum vessel, where lower ports are important appendices that house the Metal Foil Pumps and the Linear Diffusions Pumps as major components of the vacuum pumping and fuel processing systems (the so-called KALPUREX-process).
For the design development of the vacuum pumping system - especially for the design of the lower ports - two issues have to be addressed: First, the whole vacuum pumping equipment installed inside the lower ports has to be handled remotely. This requirement has a strong effect on the design of the vacuum pumps, on the other components installed inside the ports and on the design of the remote maintenance systems.
Secondly, the interspace design and the distance between vacuum pump inlet and the sub-divertor region has a strong influence on the conductance and, thus, on the required size of the pumps and/or the achievable pressure in the sub-divertor region. Based on the preliminary design elaborated before, a study has been carried out using the Monte-Carlo code ProVac3D (i) to quantify the influence of different distances between sub-divertor region and pump inlet and (ii) to investigate the effect of using Metal Foil Pumps with different length over diameter ratios (a requirement for gas separation in the Direct Internal Recycling concept) on the achievable pumping speed.
This paper concludes with a preliminary design of the vacuum pumping systems installed inside the lower ports. It shows the limitations (design as well as performance) of such a fully integrated vacuum pumping system and states clearly where special attention is required in order to come up with a sound pre-conceptional design for DEMO in 2020.

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P2.141 Design and R&D of the second tungsten divertor of EAST

The EAST superconducting tokamak upper divertor had been updated to tungsten divertor. Based on the tungsten divertor operation 10MW/m^2 heat load can be exhausted. Long pules (100s) H mode plasma was obtained. The lower divertor of EAST is still carbon plasma facing material. Upgrade the divertor to tungten the EAST will be full metal first wall with tungten for upper and lower divertor and molybdenum for other first wall. New geometry of lower divertor has bee optimized both for ITER like plasma donfiguration and semi snow-flake plasma configuration. Based on full metal plasma facing materil the EAST is capable of make more contribut for ITER and try to achieve ITER lever (400s) H mode plasma operation. The second tungsten divertor engineering design has been completed and R&D with different fromITER monoblock targets structure is in progress. It is expecte achieve 15-20MW/m^2 heat load exhaust ability. Tungsten slices brazing technology and tungsten CVD technology is studying for divertor lower heat load area. New NEG pumping system with ZAO gas absorb material is under study to be used for divertor pumping. The tungsten divertor is expect install in EAST from begining of 2019.

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P2.187 Experimental investigation on HCLL-TBS In-box LOCA

The experimental facility THALLIUM was designed and installed at ENEA C.R. Brasimone to investigate the consequence of a HCLL-TBS (Helium Cooled Lithium Lead) In box LOCA, ensuring a good level of geometrical relevance with HCLL-TBM. Within the framework of the contractual activities agreed with Fusion for Energy, a first experimental campaign was carried out in HCLL-TBS relevant conditions. The experiments simulated a rupture in a cooling plate of the HCLL-TBM with the aim to study the In-box LOCA transient. Thereafter, a new experimental campaign was developed in order to evaluate the effects of a larger He injection into the HCLL-TBM mock-up by installing an injection valve with a bigger orifice. The main aim of the experimental activity is the analysis of the accidental transient with an He injection representative of the worst conceivable accidental condition in the HCLL-TBM. The experimental campaign followed the same experimental procedure of the first one, freezing the parameters that were varied during the first campaign and instead changing the injection pressure in the range 50-70 bar and the injection time. As in the first campaign, the pressure wave evolution in different points of the facility and the effectiveness of the mitigation strategy for HCLL-TBS are the most important outcomes of the experiments. A different behavior with respect to the first campaign was observed. In particular, the first pressure peaks are smaller as a percentage of the injection pressure and the pressure is increasing during the whole transient. The relief valve on the expansion tank drives the transient, confirming the importance of the design of this component to mitigate the effects of an In-box LOCA. The outcomes of this experimental investigation are not only supporting the design of HCLL-TBS but also provide important data in view of the conceptual design of HCLL breeding blanket for DEMO.

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P2.144 FE Analyses of Eddy Currents in W7-X Plasma Vessel

Wendelstein 7-X (W7-X) is a fivefold optimized stellarator in operation in Greifswald, Germany. W7-X Plasma Vessel (PV) consists of five modules made with 17mm thick steel and having 254 openings

for ports, necessary for cooling, heating and diagnostic purposes. Both ports number and their structures are different in each PV module.
During rare plasma disruption, the plasma bootstrap current rapidly vanishes and consequently eddy currents are generated in all electrically conducting components in plasma vicinity. The evaluation and analysis of these eddy currents are of major interest, because of the electromagnetic loads resulting from current interaction with the magnetic fields created by the magnets system and the plasma. This paper focuses on the evaluation of the eddy currents induced in the W7-X PV components due to the exponential decay of the bootstrap current in the plasma (time constant 140ms). This task was accomplished by means of a PV 3D FE model. Due to the complexity of the geometry of the structure analysed, special emphasis was put on modelling issues such as elements type, mesh density and excitation representation. In first phase of the activity, the PV structure was modelled without ports, which were included in a subsequent phase. A procedure to allow quick and easy introduction or exclusion of ports from EM analyses was devised and implemented. In this way, the influence of these structures on the eddy currents could be evaluated efficiently.
Peak value of total eddy current running around the PV structure in absence of port structures turns out to be in the order of 20kA. Furthermore, results indicate that the presence of the ports leads to a relatively minor decrease of this value, in the order of 5%.
The paper presents also few examples of the PV FE model re-use for the analysis of W7-X in-vessel components.

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P2.146 Kinematic calibration for a hybrid redundant robot based on Artificial Bee Colony algorithm

In the ITER or the future DEMO fusion reactors, due to the neutron activation, the remote handling tasks such as inspection, repair and/or maintenance of in-vessel and ex-vessel components must be carried out using a wide variety of special tailored manipulators. In order to adapt to the complex environment, the accuracy of the manipulators is necessary to be improved. The kinematic calibration for a 10-DOF hybrid redundant serial-parallel robot, which is designed for ITER vacuum vessel remote maintenance, is presented in this paper. As the Kinematics of hybrid serial-parallel robot is complicated both in forward and inverse solutions, and in addition, the error model is high nonlinear, for the calibration the kinematics and error models of the hybrid robot based on the Product of Exponential (POE) formula are created.
In this paper, the artificial bee colony (ABC) optimization algorithm is used to identify the forward and inverse kinematic parameters and renew the error model. The ABC algorithm is a relatively new optimization technique that simulates the behavior of a honeybee swarm. It has the advantages of fewer control parameters and good exploration. The parameters in POE error estimation functions are evaluated by the ABC algorithm. Therefore, the assumption error parameters in the nonlinear model are identified, and the position and orientation of the end-effector are updated to more accurate values. Finally the accuracy of the end-effector is effectively improved. The methods presents in this paper could be extrapolated to the any calibration of remote handling manipulators in ITER or the future DEMO.

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P2.154 Welding analysis for the 1/16 VV sector of CFETR mock-up

This paper mainly analyzes the process of the 1/16 sector vacuum vessel (VV) welding from two 1/32 VV sectors in the 1/8 VV sector research and development project of China Fusion Engineering Test Reactor (CFETR). In the numerical simulation, the inherent strain method was applied to analyze

the welding deformation and shrinkage of 1/16 VV sector with and without welding tools respectively. What's more, it predicts the deformation trend and magnitude, which are used to compensate welding shrinkage. In the actual welding process, the welding tools and compensation are adopted, the deformation are measured through controlling points. The results show that the welding tools are effective in controlling deformation and improving the manufacturing accuracy, the trend and magnitude are consistent to the analysis result before removing the welding tools.

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P2.149 Manufacturing Study of Lower Cryostat Thermal Shield Cylinder Component for ITER Tokamak

This paper describes the manufacturing study of ITER Lower Cryostat Thermal Shield (LCTS) cylinder components, which were delivered to ITER site. Fabrication of LCTS cylinder had been proceeded according to the following processes: 1) plate cutting, 2) shell to flange welding, 3) cooling pipe welding, 4) flange final machining, 5) pre-assembly of 60 degree sector, 6) silver coating, 7) final acceptance test. All LCTS cylinder 20 degree sectors will be assembled at the ITER site by the flange joints, which are welded to the shells. Reliability of cooling pipe welding is checked by two inspection methods: endoscopy and leak test. Pre-assembly of three 20 degree sectors is one of the important step to ensure dimensional tolerance and its process is described in details. Dimensional inspection is performed for the pre-assembled 60 degree sector by a laser tracker. Flow test result is discussed for the cooling pipe routing of single sector of LCTS cylinder. Silver coating qualification is carried out using LCTS cylinder mock-up to check operating conditions of the electroplating and its coating jig design. Technical efforts to improve silver coating quality are also presented. Coating thickness and emissivity are measured for the test coupons and their results are summarized for typical LCTS cylinder sectors. Finally, packing design of LCTS cylinder for delivery is briefly described in this paper.

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P2.163 Conceptual design of the Enhanced Coolant Purification Systems for the European HCLL and HCPB Test Blanket Modules

The Coolant Purification Systems, together with the Tritium Extraction System (TES), the Tritium Removal System (TRS) and the two Helium Cooling Systems (HCSs) belong to the ancillary systems of Helium Cooled Lead Lithium (HCLL) and Helium Cooled Pebble Bed (HCPB) Test Blanket Modules (TBMs) which are currently in the preliminary design phase in view of their installation and operation in ITER.
For both European concepts of TBMs, the CPS implements two functions: the first one is the extraction and concentration of the tritium permeated from the TBM modules into the primary cooling circuit; the second one is to control the chemistry of helium primary coolant by removing the impurities from the gas and keeping slight oxidizing conditions in order to promote the formation of natural oxide tritium permeation barriers on the cooling plates of the TBM. The previous conceptual design of this ancillary system has been deeply analyzed during the HCLL and HCPB-TBS (Test Blanket System) Conceptual Design Review (CDR) in 2015.
During CDR it was recognized the need of reduce the tritium permeation into the PC#16 of ITER. Since a significant contribution comes from the HCLL-HCS piping, it was highlighted the need to strongly reduce the tritium permeation rate from the HCS piping.
To achieve this and, then, to lower the tritium partial pressure in the HCS in normal operation, the helium flow-rate treated by CPS has been increased of almost one order of magnitude
In 2017, to fully satisfy the CDR outcomes and the new design requirements/specifications requested

by Fusion for Energy (F4E, the European Domestic Agency for ITER), ENEA performed the conceptual design of the Enhanced Coolant Purification Systems.
This paper presents in detail the current design baseline of the "Enhanced" Coolant Purification Systems, focusing on design requirements, assumptions, selection of technologies and preliminary component sizing.

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P2.150 Remote diagnostics application software for remote handling equipment

ITER Remote Handling equipment controllers provide measurement and diagnostics data about the remote handling equipment and devices they control, about themselves and their operating environment. This information is aimed for the RH operators to reduce downtime of the Remote Handling systems by anticipating maintenance needs and failure conditions.

In this paper, the development of the Remote Diagnostics Application (RDA) software for the analysis and archiving of diagnostics data is presented. RDA aims to become one of the standard High-Level Control System applications of the ITER Remote Handling Control System.

RDA provides a basic set of diagnostics tools for the ITER RH operators that can be extended for specific RH equipment needs. For that purpose, RDA facilitates runtime incorporation of LabView Visual Instruments addressing specific diagnostics cases not presumed at the time of RDA programming. This has been achieved by a three-layered architecture: RDA Framework, its Diagnostics Workbenches and their Diagnostics Primitives. The RDA Framework incorporates a user interface that can load one or several special diagnostics cases implemented as custom Diagnostics Workbenches with their own GUI layout and custom or default Diagnostics Primitives, which can be rules, analysis functions, filters, etc.

RDA has the following features by default:
· data collection from RH equipment controllers into 8-hours local cache. RDA communicates with the equipment controllers through the Controller Interface Protocol (CIP)
· time domain plotting of data (trends)
· frequency domain analysis (spectra)
· amplitude domain analysis (histograms)
· scatter plots
· cross-correlation plots
· archiving and retrieval of history data (using HDF5 files).

As a result, RDA enables full exploitation of the LabView Toolkits and Modules and facilitates the implementation of complex and dedicated diagnostics and prognostics for the RH operators to:
· monitor performance data;
· run diagnostics tests and rules on equipment systems;
· analyse historical data.

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P2.151 Initial integration concept of the DEMO lower horizontal port

The realization of a Demonstration Fusion Power Reactor (DEMO) to follow ITER, with the capability of generating several hundred MW of net electricity and operating with a closed fuel-cycle is viewed by Europe and many of the nations engaged in the construction of ITER as the remaining crucial step towards the exploitation of fusion power. The DEMO machine has three main entrance

levels to the plasma chamber. According to the current DEMO reference configuration the vacuum vessel (VV) has 16 vertical upper, horizontal equatorial, and horizontal lower ports, respectively.
This article introduces the initial integration concept of the lower port that has been developed considering the external space constraints, the neutron shielding requirements of the superconducting coils, and its functions. These consist in supporting the VV, hosting different systems in particular the torus vacuum pump and feeding pipes of in-vessel components (IVCs), and to allow for divertor remote maintenance. The size and position of the lower port are constrained by the adjacent TF and PF coils. At the same time the lower port drives the layout of the cryostat and the tokamak building.

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P2.159 Electromagnetic Analyses and One-Dimensional Modeling of Piping Applied to Optimization of ITER Blanket Manifold Design

A significant analysis effort was undertaken to address the challenging ITER Blanket Manifold design requirements resulting from electromagnetic (EM) major disruptions and vertical displacements events in a very demanding neutronic environment.
The effort was focused on maintaining the structural integrity of the component itself and minimizing the loads transferred to the Vacuum Vessel to meet the requirements for the categorization of the welding between the Blanket Manifold supports rails and the Vacuum Vessel.
The complexity of the component, consisting of 360 cooling circuits (and their supports) attached to the rear side of 440 Blanket Modules and hydraulically connected to the Blanket Modules through a Coaxial Connector, required a detailed study of the current paths involving the pipes, the supports and the Vacuum Vessel, to determine the layout of electrical connections and enable the minimization of electromagnetic loads.
To this aim, a dedicated EM "wire element" was developed and validated for piping modelling. The "wire element" assimilates the pipe to a 1-d structure with equivalent electromagnetic properties, being able to represent accurately all the electromagnetic phenomena taking place in the pipe itself (i.e. both its inductive and its resistive behaviour). The use of 1d-wire elements was necessary in view of the large number of EM load cases to be run and of the complexity of the toroidal and poloidal paths of the pipes, which would have made a full 3D modelling unmanageable.
The EM analysis campaign performed led to the conception of an optimized layout of the electrical connections within the Blanket Manifold and between Blanket Manifold and Vacuum Vessel, which provided a simpler and much faster means to analyse different scenarios in order to minimize the loads transferred to the Vacuum Vessel and maintain the structural integrity of the Manifold.

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P2.024 Experimental Studies on Arc Chamber Failure Mechanisms on DIII-D Neutral Beam System

Neutral Beam Injection (NBI) is used for non-inductive heating, current drive, fueling and diagnostics in most major magnetic confinement fusion devices. The DIII-D device comprises eight NBI ion sources based on the US Common Long Pulse Source (CLPS), with a total output power of 20 MW.
Here we report on efforts to improve performance and longevity of the NBI system by initiating a R&D program aimed at studying the most common failure mechanisms for the ion sources. Toward this end, a filament driven plasma chamber has been constructed that attains plasma parameters similar to the arc chamber of a NBI ion source. This Miniature Arc Chamber Experiment (MACE) has a diagnostic suite that includes Langmuir probes, spectroscopy, infrared imaging and mass spectroscopy.

A report on investigations into two common failure mechanisms is presented here: Firstly, a failure mechanism observed during helium beam operations on DIII-D that results in electrical breakdown of the insulation material that separates the filament plates from the anode. The fault is reproduced in MACE and the proposals for amelioration of the issue will be discussed. Secondly, the failure of the water-cooled Langmuir probes that are necessary for beam current control will be discussed. The original probe design requires a complicated manufacturing technique involving several intricate brazes. When this component fails, it can produce a water leak that impacts operational availability. On MACE, we are testing a redesign of this component to provide a more robust solution.

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P2.189 The plan of continuous tritium recovery campaign by PbLi droplets in vacuum

Practical plan of "continuous tritium recovery by PbLi droplets in vacuum" campaign is introduced. This campaign aims to verify the viability of tritium recovery method by PbLi droplets in vacuum as a prototype design level. Following verifications are to be performed. 1) To verify the steady state extraction efficiency of tritium from a vacuum sieve tray for 1.1 mm diameter droplet. Predicted extraction efficiency from the previous experiments is higher than 80%. 2) To verify the mutual interference effect by multiple nozzles on degrading extraction efficiency. Extraction efficiencies by simultaneous 7 nozzles ejection and by 19 nozzles are verified. No major influence, even though, is predicted by previous analysis. 3) To verify the reliability of the vacuum sieve tray in continuous operation. At least 48 hours of non-stop operation is primary goal of this trial. A dedicated extraction system is designed, fabricated and installed into the "Oroshhi-2; PbLi experimental loop at National Institute for Fusion Science (NIFS)", Toki, Japan. Followed by the stand-alone test with short flowing period at Kyoto University, the experiment will be installed in the Oroshhi-2 in late 2018. Deuterium, instead of tritium, is used for the experiment. Extraction efficiency is calculated by comparing the deuterium concentration of before-drop and after-drop by measuring the permeating amount through the test tube wall. The amount of released deuterium is also measured and verified with the before-and-after concentration change. PbLi temperature is between 400℃ and 500℃, and the flow velocity at the nozzle is 3 m s-1. Initial experiment is scheduled in March 2019.

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P2.029 Review of the JET ILA Scattering-Matrix Arc Detection System

Arc detection is an essential protection system for high power RF systems. It is commonly realised by monitoring the Voltage Standing Wave Ratio (VSWR) in the transmission lines. The JET ILA is a load tolerant ICRF antenna composed of 8 short straps grouped in 4 Resonant Double Loops (RDLs). In this type of antenna, there is a low impedance section in which the standard VSWR protection is ineffective.

The Scattering-Matrix Arc Detection (SMAD) was proposed and installed on JET [1] to protect the low impedance section around the T-junction against arcing. It is based on a consistency check of the RF signals around this section using a table of correlation coefficients obtained from RF modelling. This contribution reviews the SMAD protection system and its recent improvements, the conditions in which it is essential to protect the antenna during operation, the commissioning of the system and its sensitivity to the input signal levels and accuracy.

The SMAD error remains small in the full ILA operation frequency range (28-51MHz) during operation on L-mode and H-mode plasmas, showing that both the RF model of the antenna circuit and the

measurements are sufficiently accurate for protection purposes. Moreover, due to its insensitivity to the RF coupling properties and the fast (2μs) FPGA error calculation, the SMAD is suitable for detecting arcs during the ELM cycles.

The time delay to issue a SMAD trip is typically set to 6-8μs (3-4 FPGA cycles), which is confirmed by scope measurements and fast data signals analysis. The time delay between the trip signal and the effective removal of the power in the generator output transmission line is measured to be about 25μs, which is within the general protection specifications of the ICRH system in JET.

[1] M. Vrancken et al., Fus. Eng. Des. 84 (2009) 1953-1960.

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P2.157 Operation of probe heads on the Multi-Purpose-Manipulator at W7-X

The Multi-Purpose-Manipulator (MPM) has been operated, as a versatile carrier system for probes, since the first campaign OP. 1.1 in 2015 at Wendelstein 7X (W-7X). The combined probe, a combination of Langmuir, Mach and magnetic probes, was used. For the second campaign OP. 1.2a in 2017, with an island divertor, an upgraded combined probe, a fluctuation probe, a retarding field analyzer (RFA), a Mach probe array, a laser blow off target and a material probe holder were added to the range of plasma edge diagnostics. In addition, a system for gas injection in the form of a dedicated head and an additional port on existing probes for fueling and impurity seeding experiments was utilized.
The new diagnostic probe heads were designed to address specific tasks, such as measuring the edge fluctuations, the ion temperature and the radial Mach number profiles. The minimal setup, for probes that were plunged with a fast stroke, was the measurement of the electron temperature and density and the radial electric field. The probes were able to measure in the scrape off layer up to the last closed flux surface. By combining the measurements from all probes, the full picture of the profiles of the electron temperature and density for all configurations, that were used during the second campaign, can be studied. Not in the same position, but in the same flux tube or comparable radial range are other diagnostics like the divertor Langmuir probes, the poloidal correlation reflectometer and the infrared cameras.
This contribution will use the overlap of diagnostics, that allowed for a more complete characterization of the edge plasma parameters during the campaign, by comparing the probe results with each other and complementary edge diagnostics.

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P2.158 Upper port #02 and #08 structure integrity report

This presentation shows results of calculations of the Upper port plug (UPP) and ex-vessel components of ITER upper ports №2 and №8. Detailed finite-element models of modernized UPP construction were developed taking into account nonlinear contact interaction between the diagnostic shield module and the UPP structure. In the structural analysis under design loads the stress-strain state (SSS) taking into account the bolts pretension was obtained. Besides, in the new FE model pipelines structure of the cooling system was considered.
In thermal analysis of the UPP for baking and normal operation modes the neutron analysis data was taken into account. The UPP cooling systems' hydrodynamic characteristics were calculated including the flow-pressure characteristics. Subsequently, the whole UPP construction flow and pressure characteristics were evaluated taking into account the presence of throttling devices in the UPP hydraulic circuit.
Seismic analysis was carried out with the linear spectral method based on the response spectra re-

ceived from the UPP Generic model analysis. The input response spectra were taken at the points of the port attachment to the vacuum vessel. The spectral power density analysis of the interface effects was carried out in Ansys. Results were considered at the anchorage points of the UPP to the port and converted to response spectra for linear spectral analysis of the UPP construction. Seismic analysis of the ex-vessel components was carried out based on the response spectra at the UP point of the ITER building.
Ponderomotive force distribution in the UPP and ex-vessel components under different scenarios of plasma disruption was calculated. Electromagnetic model was significantly refined. So, it was possible to analyze each part of the UPP construction separately and determine the extremely loads acting on it. Based on the electromagnetic analysis data the structural analysis of the UPP and ex-vessel components under electromagnetic loads was carried out.

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P2.173 Fuel Retention Diagnostic Setup (FREDIS) for desorption of beryllium and tritium containing samples

In fusion devices, the retention of the fusion fuel deuterium (D) and tritium (T) in plasma-facing components (PFCs) is a major concern. Measurement of the hydrogen isotope content in PFCs and test samples gives insight into the retention physics.
In FREDIS, Thermal Desorption Spectrometry (TDS) is performed in an evacuated (p < 1E-8 hPa) quartz tube (Ø52 mm), where samples are heated by 6 infrared lamps up to 1433 K with linear temperature ramps of up to 1.67 K/s. The desorbed gases are detected up to 100 amu with a double-QMS (Quadrupole Mass Spectrometer) that can distinguish between helium and D2 and uses an innovative differential pumping system.
In a connected vacuum chamber (Ø250 mm), a Ø3 mm spot can be heated on the sample surface by a high energy Nd:YAG laser pulse (E < 100 J) within milliseconds (t < 20 ms) to several thousand degrees, ensuring complete fuel desorption within the heated volume [1]. This method of Laser-Induced Desorption (LID) can also be applied inside the fusion chamber and is planned as in situ retention diagnostic for ITER [2]. In FREDIS, LID is used as ex situ analysis method using the same double-QMS for absolute quantification. In this contribution we present the specifications of FREDIS and compare TDS and LID.
Moreover, FREDIS is designed to handle beryllium by means of glove boxes and in the future also tritium using a tritium trap. Therefore, FREDIS is located in a radiation controlled area – the High temperature Materials Laboratory (HML) – in the Forschungszentrum Jülich. FREDIS is capable of analysing beryllium- and T-containing samples from JET and potentially ITER.

This work is supported by: EUROfusion EEG-2017/05 (AWP17-EEG-FZJ/Zlobinski)

[1] M. Zlobinski, Jülich: Forschungszentrum, Zentralbibliothek (2016), Juel-4396, hdl.handle.net/2128/12453
[2] G. De Temmerman, et al., Nuclear Materials and Energy 12 (2017) 267-272, doi:10.1016/j.nme.2016.10.016

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P2.030 The WEST plasma control system: Integration commissioning and operation on the first experimental campaigns

The WEST tokamak is aiming at testing ITER like divertor component. This requires to address new control challenges like X-point configuration magnetic control or heat loads control in metallic environment and event handling challenges to sustain long duration H-mode that are in line with ITER needs.
To address these requirements, a new Plasma Control System (PCS) has been built using a generic

version of the DCS (Discharge Control System) Real-Time (RT) framework that is currently used on ASDEX-Upgrade and offers enough flexibility to be adapted to any tokamak. Based on a segmented approach of the plasma discharge, DCS is now the central part of the WEST plasma control system, managing the different actuators and reading the RT data from a large set of diagnostics. In order to reach this goal, several modules have been developed for the synchronization and the communication between DCS and the WEST CODAC infrastructure. The WEST controllers' layout has been built using Matlab Simulink while the RT source code has been generated with the Simulink coder toolbox. This approach allows asserting the controllers' performance by coupling them to a tokamak simulator prior to the commissioning phase.
This contribution will start by summarizing the different concepts used to build the WEST Plasma Control System. The details of the modules developed to integrate DCS into the global WEST CODAC infrastructure will be discussed. An overview of the different plasma controllers will be also presented. The last part of the paper will focus on the results obtained so far highlighting the performance of the system and on the analysis tools used during operation. We will also discuss the ability of the WEST PCS concepts to deal with the machine protection issues.

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P2.164 ARC reactor: Activation analysis of the liquid blanket and structural materials for the vessel

Nowadays, Fusion Energy is one of the most important sources under study. During the last years, different designs of fusion reactor were considered. At the MIT, an innovative design was created: ARC, the Affordable Robust Compact reactor. It takes advantage of the innovative aspect of recent progress in fusion technology, such as High Temperature Superconductors, that permit to decrease the dimension of the machine, reaching at the same time high magnetic fields.
Our main goal is the low-activation analysis of possible structural materials for the vacuum vessel, which is designed as a single-piece placed between the first-wall and the tank that contains the breeding blanket. Due to its position, the vacuum vessel is subject to high neutron flux, which can activate it and cause the reduction of the component lifetime and decommissioning problems. The activation analysis was done also for the liquid breeder FLiBe, compared with Lithium-Lead. Codes used for the low-activation analysis were MCNP and FISPACT-II. The first one is based on a neutronic model and for each component a certain neutron flux is evaluated. For FISPACT-II, the main input is the composition of the analyzed material, the neutron flux and the irradiation time. Results from FISPACT-II are the time behavior of Specific Activity, contact Dose Rate, and Decay Heat. To choose the best structural material for the vacuum vessel, both mechanical and low-activation properties were considered. Vanadium alloys turn out to be one of the best alternatives to the present material, Inconel-718. Finally, Isotopic Tailoring and Elemental substitution methods were applied. Here, the composition of each alloy is analyzed and critical isotopes or elements are eliminated or reduced. After the modifications, new simulations are done, and those leading to significant improvements in the final results are highlighted.

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P2.169 Feasibility study for long-lived fission products transmutation using fusion reactors

Long-lived fission products (LLFPs) are one major factor of the radioactivity and decay heat of high level wastes produced by light water reactors. Transmutation of LLFPs into non-radioactive or short-lived nuclides is an efficient way to reduce the amount of high level waste. Earlier studies have proposed to use a part of the breeding blanket of a fusion reactor to transmute LLFPs. The earlier studies focused mainly on the amount of LLFPs that can be transmuted. Loading LLFPs in the limited space,

however, affects the characteristics of the blanket system, such as a decrease of the tritium breeding ratio (TBR) and nuclear heating.
In this study, the feasibility of LLFPs transmutation using a fusion reactor was discussed. More specifically, the amounts of LLFPs transmuted, the TBR, and the nuclear heat generated were evaluated by a numerical analysis. Neutron transport and burn-up analyses were conducted by using MVP-2.0 and MVP-BURN. A simple model of LHD type helical fusion reactor FFHR-d1 was used to obtain the amount of the target nuclide reduction and the TBR. One of the target nuclides, Se-79, Zr-93, Tc-99, Pd-107, I-129, or Cs-135 were loaded into the breeding blanket. As a result, when the amount loaded was 10% of the breeding blanket volumetric ratio, the effective half-life was shorter than the decay half-life. By an evaluation using a support factor, which refers to the number of million kW class light water reactors that can be covered by the transmutation reactor in one year. For several nuclides the support factor was less than 1. This means the amount of LLFPs transmuted needs to be raised by changing conditions. Other evaluations of the TBR and amount of nuclear heat generated were conducted and the feasibility was discussed.

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P2.166 Commissioning of Multi-Nozzle Vacuum Sieve Tray at the Tritium Laboratory Karlsruhe

In order to achieve tritium self-sufficiency of fusion reactors, tritium will be generated in breeding blankets by neutron bombardment of lithium and then will be extracted to refuel the plasma. The Vacuum Sieve Tray (VST) was proposed to ensure tritium extraction from liquid breeding blankets, composed of lead-lithium. It consists in letting the liquid metal fall through submillimeter diameter nozzles in a chamber maintained under vacuum. PbLi liquid jets are first formed and then break into droplets, in which the tritium dissolved in atomic form is transported towards the surface to form gaseous T¬2 that is extracted.

The multi-nozzle VST setup at the Tritium Laboratory Karlsruhe (TLK) is designed to study the scalability of the VST technique with a setup operated with deuterium. This setup is fully assembled with tritium-compatible components to serve also as a preliminary deuterium/lead-lithium facility before performing experiments with tritium with a dedicated setup at the TLK.

The presentation will include a short introduction to the facility and will focus on the commissioning to ensure safety and prepare the experimental campaign (experiments and calibration tests in order to obtain accurate results and to benchmark a hydrodynamic simulation code previously developed). The experimental strategy will also be presented.

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P2.167 Numerical study and experimental verification of protium permeation through Pd/Ag membranes for fusion applications

Pd/Ag membranes are one of the reference technologies for the fuel cycle of deuterium-tritium fusion machines. This technology is proposed to be implemented in tritium recovery systems, due to their exclusive selectivity towards molecular hydrogen isotopes. For instance, these membranes are proposed to process and separate Q2 (Q = H, D, T) species from impurities (e.g., inert gases) coming from the plasma exhaust. In addition, the combination of Pd/Ag membranes with catalyst beds allow the decontamination of tritiated compounds (like Q2O, CQ4, NQ3, etc.) as a result of permeation and catalytic reactions. For these peculiarities, the Pd/Ag membrane reactor was proposed to continuously separate tritiated species (mainly HT and HTO) from helium in the tritium extraction system of the solid blanket.

ENEA has recently developed Pd/Ag single-tube membrane reactors, which have been tested for the separation of H2/He mixtures (by permeation) and decontamination of D2O/He streams (permeation plus reaction). In view of up-scaling this technology to a DEMO-relevant case, a one-dimensional simulation code was developed to first predict H2 permeation efficiencies. In this contribution, a numerical code which computes the permeation efficiency of protium through Pd/Ag membranes at different conditions is presented. Geometrical (i.e., length, inner diameter, thickness), operational (i.e., pressures, temperatures) and membrane (i.e., permeability) parameters, as well as the initial feeding gas compositions, are given as input. A sensitivity analysis is presented for an understanding of the most impacting parameters in the membrane's performance. Furthermore, the very good correlation between numerical outcomes with actual experimental results obtained at ENEA is highlighted and discussed.

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P2.168 Synthesis of SiC films as tritium permeation barriers with high growth rate using a helicon wave plasma technique

Tritium permeation loss in the fusion reactor is an important issue. Silicon carbide (SiC) is considered as an important material for Tritium permeation barriers due to its excellent properties (including low diffusivity). Steady-state and high-flux helicon-wave excited Ar/CH4/SiH4 plasma were used to synthesis SiC film onto 316L stainless steel. The surface profile and the thickness of the films were characterized by scanning electron microscopy. X-ray photoelectron spectroscopy (XPS), Infrared absorption spectroscopy and Raman spectroscopy were used to characterize the composition and microstructure. Plasma parameters such as electron density (ne), electron temperature (Te) are measured by optical emission spectroscopy and Langmuir probe. The growth rate of SiC film in different discharge mode is also investigated. The growth rate reaches a maximum value of about 3nm/s, which is attributed to the higher plasma density during the helicon wave plasma discharge. The experiments open up a new way for achieving high growth rate of SiC films at lower temperature.
Keywords: helicon-wave excited plasma, Tritium Permeation Barrier, SiC film.
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P2.033 Actuator management development on ASDEX-Upgrade

The future AUG control research is expected to cope with a large number of control tasks using a limited number of actuators in high performance regime. Essential part of a control system for future tokamaks is an intelligent actuator management that will be responsible for allocating the most convenient actuators to the control tasks of the highest importance.

Activities in this field have been started at DIII-D, AUG and TCV. The first version of the actuator management at AUG was developed for MHD control and impurity accumulation preemption using ECRH with fast mirrors [Rapson et al, FED 2017]. The next step will be to develop actuator management routines applicable to all heating actuators available at AUG: ECRH, NBI, and ICRH. The first application of this will be in beta control both using NBI and ECRH (so far it has been possible only by NBI) and extension of the applicability of the Te profile controller, which currently uses one gyrotron per channel and suffers from saturation problems. The actuator management will allow for

more flexible allocation of actuators to Discharge Control System controllers and for the combination of different actuator types. In later stages, a plasma control supervisor capable of repurposing actuator for different control tasks will be developed.

This contribution will discuss the techniques that allow for reaching such a flexibility. The first experimental results in beta and Te profile control using the actuator management will be reported. Also, the first concept of the real time plasma control supervisor will be presented.

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P2.170 Evolution of constrained beryllium pebble bed mock-ups neutron-irradiated up to 6000 appm helium production

In the European Helium-Cooled Pebble Bed (HCPB) concept of the DEMO blanket, beryllium pebbles with a diameter of 1 mm are planned to be used as neutron multiplier. A study pebble bed mock-up behavior under high-dose neutron irradiation at the HCPB relevant temperatures in a material testing nuclear reactor should provide an essential database for blanket designers.
Beryllium pebbles with diameters of 0.5, 1, and 2 mm produced by the rotating electrode method were irradiated in the HIDOBE-02 experiment in the HFR, Petten at temperatures of 370-650 °C up to damage doses of 21-38 dpa, corresponding to generation of 3600-6000 appm helium. To simulate thermo-mechanical, swelling and creep issues of the HCPB blanket, the constrained beryllium pebble bed mock-ups with a diameter of Ø14.9 mm and heights of 32 and 45.5 mm were used. Along with the mock-ups, unconstrained beryllium pebble stacks were also irradiated. Post-irradiation examinations included optical microscopy of cross sections of the pebble bed mock-ups and unconstrained pebbles, as well as micro-hardness measurements. This contribution is mainly focused on the behavior of constrained pebble beds.
Neutron irradiation resulted in evolution of the pebble microstructure depending on the irradiation temperature. For unconstrained pebbles, formation of gas bubbles becomes more pronounced above 560 °C. The constrained pebbles obey more developed sub-grain structure retarding bubble coalescence. Sintering of the pebbles in the pebble beds is especially noticeable after irradiation at the highest temperature of 600 °C. For 1 mm constrained beryllium pebbles, micro-hardness reduces from HV0.4 234 at a temperature of 387 °C to HV0.4 164 at the highest irradiation temperature of 600 °C.
The analysis of the obtained results provides deeper understanding of microstructure evolution in the constrained beryllium pebble beds and unconstrained pebbles after high-dose neutron irradiation.

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P2.221 Accident analysis with MELCOR-fusion code for DONES lithium loop and accelerator

Safety assessment is a key issue for the licensing of DONES facility, the DEMO-Oriented Neutron Source. A first phase of the safety assessment include Failure Mode Analyses of systems to identify postulated initiating events (PIEs), while deterministic accident analyses are lately performed to estimate source terms in radiological hazards. In addition, the deterministic analyses are also the basis to identify safety class systems and components, and to establish the operational range of system parameters as well as the actuation of the protection system.

In this paper two accident scenarios, previously identified as PIEs, have been simulated with the fusion version of the MELCOR 1.8.6 code. A loss of flow (LOFA) in the lithium loop due to the trip of the electromagnetic pump (EMP) and a loss of vacuum (LOVA) in the accelerator beam duct due to containment rupture have been studied to provide a preliminary estimation of the available actuation time for detection and mitigation systems with a particular focus on the beam shutdown system and isolation of the vacuum duct.

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P2.172 The production and molecular occurrence of radiotoxic Po-210 in liquid Pb-Li tritium breeding blankets

To be accepted in the future energy landscape, fusion reactors must be inherently safe by design. An unresolved safety issue is the undesired production of highly radiotoxic Po-210 in the liquid Pb-Li eutectic used in many breeding blanket concepts. Po-210 is the end product of consequent neutron captures and beta decays, initiated by a neutron capture by Pb-208.

Po-210 is an intense alpha emitter, having a median lethal dose of only 0.089 micrograms. To ensure the safe operation of a fusion reactor, the inventory build-up of Po-210 should be monitored over the lifetime of the reactor. Therefore, 3D Monte Carlo neutron transport calculations were performed using the MCNP code for realistic DEMO reactor models, developped by the Power Plant Physics and Technology programme. Two blanket concepts based on the use of Pb-Li were considered: the Helium Cooled Lithium Lead and the Water Cooled Lithium Lead blankets. The obtained neutron fluxes were coupled to the FISPACT-II code to determine the time-dependent Po-210 inventory. Both conservative and more realistic estimations were determined by using different DEMO models, nuclear data libraries and initial Bi impurities in the Pb-Li.

If the Po-210 inventory becomes too high, filtering techniques must be applied. To devise efficient filtering systems, the molecular composition and aggregation state of the Po-210 have to be determined. We focused on the properties of Po-containing gaseous molecules, as these can escape from the reactor most easily. As experiments with volatile Po species are subject to severe safety restrictions, few experimental data is available. An alternative quantum chemistry approach with the MOLCAS code was used to predict the stability of all relevant diatomic Po-containing molecules. Using experimental data on lighter analogue molecules, the reliability of the method was assessed. Finally, the relative occurrences of the different Po species for realistic operational conditions was estimated.

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P2.190 Li-rod structure in high-temperature gas-cooled reactor as a tritium production device for fusion reactors

To start up an initial fusion reactor and for technical tests for tritium circulation and blanket system, it is necessary to provide sufficient amount of tritium from an outside device. Tritium production using a high-temperature gas-cooled reactor has been proposed. [1]. It was reported that 500–800 g of tritium could be produced during one year of operation using a 600 MW thermal output high-temperature gas-cooled reactor [2]. If we can keep the Li-rod temperature below 800 K during the operation, the tritium outflow from the Li rods to He coolant could be suppressed to less than 1% of the amount of the tritium produced. When we attempt to operate the reactor at a higher temperature range (e.g. 1100-1200 K) from the viewpoint of electric power generation efficiency, the outflow will be increased. It is important to devise a way to reduce the outflow to further low level.
In this paper, we assume the tritium production using HTTR (High Temperature engineering Test Reactor). Amount of tritium produced was evaluated using continuous-energy Monte Carlo transport

code MVP-BURN [3], and tritium containment using diffusion equation. Nuclear data were taken from JENDL-4.0, and tritium permeability and absorption data were taken from our experiment. The purpose of this paper is to clarify a suitable structural design of the Li-rod from the viewpoints to produce sufficiently large amount of tritium and to securely contain the tritium produced in the Li-rod during the standard reactor operation with 1100-1200 K rod temperature.

[1] H. Matsuura, et al., Nucl. Eng. Des. 243 (2012) 95.
[2] X. Yan, et al., Nucl. Eng. Des. 222 (2003) 247.
[3] Y. Nagaya, et al., JAERI 1348 (2005).

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P2.035 Unified Signal Identifier for Globally Unique Signal-Addresses

Wendelstein 7-X (W7-X) stellerator has been designed to support a long-term and continuous operation. In that concern, corresponded scientists have access on the archived data (signals) anytime-anywhere, where archived signals can be referenced via project-specific unique identifiers, referred to as signal-addresses.
At the same time, different projects in the fusion research such as W7-X and ITER use different address-schemes for addressing the archived and measured signals. Even in the same project, different development groups such as for desktop and web-applications may use different forms of addresses for referencing the same signals. However, (1) scientists in a project should be able to access an archived signal locally as well as remotely using the same signal-address and address-scheme; (2) scientists using data of different projects should be able to use the same addressing scheme for referencing the required signals. The main requirements for achieving this purpose can be summarized in two main points: (1) globally unique signal-address, independent of the storage media and location as well as the research project and organization; (2) unified address format (syntax) and semantic, as standard for all projects in the field of fusion research.
This paper represents the first step to unify the addressing schemes in fusion research for archived signals depending on experience gained and results achieved in the project W7-X and based on the well-known Uniform Resource Identifier (URI) standard. Accordingly, a novel concept is proposed for a Uniform Signal Identifier (USI) as a domain-specific address scheme for fusion research; the USI uses the URI standard as well as the addressing schemes of Java-API and Web-API of W7-X as basis. Additionally, this paper presents the USI terminology as well as syntax and semantic, suggests a system architecture and provides an integration example based on the current development of W7-X Archive.

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P2.036 The new W7-X Logbook - A Software for Effective Experiment Documentation and Collaborative Research at Wendelstein 7-X

Wendelstein 7-X (W7-X) completed its second operation phase (OP1.2a) in December 2017. A large number of diagnostics were operated in nearly 1000 experiment programs by an international research team. For the documentation of W7-X experiment programs, a new electronic logbook software was developed and eventually used for the first time in OP1.2a. The software was designed for the needs of W7-X researchers: a web-based logbook application for the whole team. For an effective documentation, greater part of the logbook content comes from automatically generated logs, complemented by users via web browser. The W7-X control software generates log entries for experiment programs during the program execution. This includes an automatic extraction of configuration information from the planned program, which is later represented as searchable tags

within the logbook. The logbook allows full-text search and range queries for numeric values: both with response times within milliseconds. Separate component logs are created in the same way for W7-X diagnostics and machine sub-systems for different use cases, e.g. experiment programs, standalone tests, or calibrations. The logbook allows adding rich-text comments to all logs, which also includes links and images. The web page of each experiment program log contains overview plots from measurement data, generated on the fly from data from the W7-X archive. For each W7-X component, the logbook also provides a separate web page, which is editable by the responsible officer. The pages contain descriptions, log overviews, and links to measurement data. Next to the website interface based on HTML5 and JavaScript, the logbook provides a RESTful web service, which allows reading and writing of logbook data as JSON. These features are making the logbook an excellent experiment journal, which was quickly adopted by the W7-X team and appreciated as a crucial tool for experiment documentation and starting point for data analysis.

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P2.176 Deuterium permeation behavior through reduced activation ferritic steel F82H under DEMO reactor blanket condition

Tritium permeation through structure materials in fusion blanket systems is a critical issue from the perspectives of fuel loss and radiological hazard. In the previous studies, detailed hydrogen isotope permeation behaviors in reduced activation ferritic/martensitic (RAFM) steels have been investigated; however, it is supposed that the surface of the RAFM steel will be oxidized under an actual DEMO reactor condition at a tritium recovery system, and then the permeation behavior will be changed. In this study, deuterium permeation through the RAFM steels heat-treated under simulated environment conditions has been investigated for more precise prediction of tritium loss at DEMO reactor blankets.
RAFM steel F82H substrates were heat-treated in helium gas flow containing 1 vol% hydrogen for 100 h at 300, 400 and 500 °C to simulate a DEMO reactor blanket condition. After surface observation and analysis for the heat-treated samples, gas-driven deuterium permeation measurements were performed. Deuterium permeation flux through the sample was detected by a quadrupole mass spectrometer with the driving pressure of 10.0–80.0 kPa in the temperature range of 250–550 °C.
An iron oxide layer was formed on the F82H surface after the heat treatment. From the results of grazing incidence X-ray diffraction analysis, the major composition of the surface oxidation layer was Fe2O3. The oxide layers became thicker as the heat-treatment temperature increased. The thickness of the oxide layers of the samples heat-treated at 300, 400 and 500 °C was approximately 50 nm, 130 nm and 5 μm, respectively. In the results of permeation tests, the iron oxide layers of all the samples decreased the deuterium permeation down to only 25%, suggesting that the structure of the iron oxide layers were not tight; therefore, the layers naturally formed under the DEMO blanket condition will not be efficient as tritium permeation barriers.

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P2.196 Corrosion resistance of alumina forming steel and ceramic materials in liquid tin

Liquid tin (Sn) is a promising coolant of liquid divertor systems due to its low vapor pressure. However, the material compatibility with structural materials is important issue for the development of the liquid divertor system. The purpose of the present study is to explore corrosion resistant materials in the liquid Sn. The corrosion tests were performed in a static Sn at 773K with various types of test samples. The test materials are reduced activation ferritic martensitic steel JLF-1 (Fe-9Cr-2W-0,1C), alumina forming steel (Fe-17.7Cr-3.3Al-0.4Si), pure tungsten (W), pure chromium (Cr) metal,

pure titanium (Ti) metal, pure zirconium (Zr) metal, silicon carbide (SiC), silicon nitride (SiN), aluminum nitride (AlN), iron oxide (Fe2O3) bulk, alumina (Al2O3) bulk and chromium oxide (Cr2O3) bulk. Some samples were tested after pre-oxidation treatment in air at 773K. The test duration was 262 hours. After the corrosion tests, the corroded surfaces of the tested samples were metallurgically analyzed by FE-SEM/EDX. The test results indicated that the alumina forming steel revealed a corrosion resistance in the liquid Sn, though the surface was partially corroded by the formation of Fe-Cr-Sn alloy. The ceramic materials also revealed a corrosion resistance in the liquid Sn. The pre-oxidation treatment could be effective to mitigate the corrosion. These results indicated that the material compatibility could be improved by the application of corrosion barriers such as oxide layers and other ceramic coatings.

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P2.178 Iron-ion irradiation effects on microstructure and deuterium permeation in yttrium oxide coating fabricated by magnetron sputtering

Tritium permeation through structural materials in a fusion reactor fuel system causes fuel inefficiency and tritium leakage to the environment. Tritium permeation barrier (TPB) has been intensively developed using ceramic coatings to establish liquid blanket concepts for several decades. In a TPB coating, not only tritium permeation reduction but also tolerance to high dose radiation is required due to a severe neutron irradiation environment in the blanket region. Yttrium oxide (Y2O3) has been recently investigated as a TPB coating material with high permeation reduction and low activation by neutron irradiation. In this study, the irradiation effect on the characteristics of Y2O3 coating has been investigated using iron-ion irradiation as a simulation of damage introduction by neutron irradiation.
Y2O3 coatings were fabricated on F82H plate substrates by reactive magnetron sputtering without substrate heating. The coated samples were annealed in vacuum (<10−6 Pa) at 600 °C for 24 h to promote crystallization of the coatings. Thereafter, the samples were irradiated by Fe ion at room temperature and 500 °C. The incident energies were 1.0, 2.8 and 6.4 MeV. The displacement damage was set at up to 0.5−20 dpa, and then cross-sectional observation using a transmission electron microscopy and deuterium permeation tests were performed for the irradiated samples.
The samples irradiated by 1.0 MeV-Fe ion up to 3, 10 and 20 dpa at room temperature showed an amorphous layer near the interface. The thickness of the amorphous layer was about 100 nm in all the irradiated samples. In addition, a microcrystal layer was confirmed on the amorphous layer in the 10 and 20-dpa-damaged samples. These layers included many iron atoms and some of them formed an iron oxide. Incident energy dependence on microstructure and deuterium permeation behavior for the samples will be included in the presentation.

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P2.179 Numerical investigation of mechanical and thermal characteristics of binary-sized pebble bed using discrete element method

The characteristic identification of the functional material is important in the performance prediction of a breeding blanket, which is one of the main components of the fusion power plant. The functional material of the solid type ceramic breeding blanket is mainly used in the form of a pebble bed, which is an aggregate group of pebbles. Various experimental methods such as laser flash, hot wire, and hot disk method have been studied to assess thermal properties of the pebble bed. In recent years, numerical techniques have been also introduced to evaluate mechanical characteristics as well as thermal characteristics of pebble beds.

In this study, the mechanical and thermal behaviors of the binary-sized pebbles in the cuboidal container under cyclic loadings are simulated considering the contact interaction of the pebbles by means of discrete element method. Through a series of numerical simulations, we investigate the effects of the size discrepancy of the binary-sized pebbles on the packing and mechanical properties of the pebble bed. The stress or contact force concentration of the pebbles is also discussed, which may lead to pebble failure. Furthermore, heat conduction between the pebbles is considered based on the packing configurations obtained from the mechanical analysis, and the thermal properties of the pebble bed are qualitatively evaluated in terms of the effective thermal conductivity of the pebble bed.

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P2.180 Discrete element code to simulate the heat transfer inside ceramic breeder pebble beds

Tritium breeder pebble beds are multiphase materials where ceramic pebbles and purge gas coexist. Therefore, the heat transfer in the bed is influenced by both solid particles and helium gas. Furthermore, due to their discrete nature, pebble beds show a complex fully coupled thermo-mechanical behaviour. Simulations carried out with the Discrete Element Method (DEM) allow evaluating the thermo-mechanical behaviour of pebble beds as a result of the interactions among particles accounting for the influence of the filling gas.

In KIT a predictive in-house thermal-DEM code was developed to evaluate the heat transfer inside packed granular assemblies. A 3D thermal network model with was simulates the interconnected particles by thermal resistors. The Smoluchowski effect was implemented for the first time in a DEM code allowing simulating the effect of the gas pressure on the effective thermal conductivity.

In this work the code was used to investigate the heat transfer inside breeder pebble beds. An assembly of packed pebbles was simulated being consistent with the Helium Cooled Pebble Bed (HCPB) test blanket concept and pebbles parameters such as: bed thickness, relevant packing factor and representative size distribution of pebbles. Power density of neutron irradiation was applied to the particles, which were packed into a box with an upper and bottom wall at 500℃ simulating the cooling plates of the HCPB test blanket module. Helium was modelled as stagnant purge gas filling the voids among particles. The temperature profile at different distances from the first wall was evaluated. Sensitivity studies were performed to investigate the influence of the gas pressure, gas type, solid material and pebble size. The influence of the cyclic compressive load was evaluated coupling the thermal with the mechanical KIT-DEM code.

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P2.205 IFMIF-DONES Systems Engineering Approach

In the framework of the EU fusion roadmap implementing activities, an accelerator-based Li(d,n) neutron source called DONES (Demo-Oriented early NEutron Source) is being designed as an essential irradiation facility for testing candidate materials for DEMO reactor and future fusion power plants. DONES facility is being developed within the EUROfusion workpackage WPENS which main objective is to be ready for IFMIF-DONES construction in 2020. Fourteen Research Units around Europe, as well as industry Third Parties, are involved working in different aspects of DONES.
Taking into account the complexity of the facility and the geographical dispersion of the partners involved, it is of a paramount importance to properly develop the DONES Systems Engineering, creating and executing interdisciplinary processes to ensure that the DONES defined objectives are reached and that the facility fulfils the expected criteria.
This paper presents the Systems Engineering processes that are being developed for the DONES

Project as an interdisciplinary approach to enable the realization of successful Structure, Systems and Components (SSCs). In order to reach this objective (defining, controlling and documenting the requirements and interfaces) a dedicated group has been set up. A matrix for the management (traceability) of the requirements is being developed and populated. Also, a specific tool is being used for the management of the project interfaces (physical, functional, etc.). These works are to be considered as part of the development phase, started as early as possible in the project and then used in further stages (design synthesis activities, SSC validation, subcontractors' control, etc.) following the Systems Engineering works.

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P2.184 Design of an industrial catalytic membrane reactor prototype for tritiated gaseous effluent treatment

In the framework of gloveboxes tritiated gaseous effluent treatment, efficiency of packed bed membrane reactors has been successfully demonstrated under lab scale. In such an intensified process, tritium from tritiated water can be recovered under the valuable Q2 form (Q = H, D or T) thanks to isotope exchange reactions on catalyst surface. In the meanwhile, the use of permselective Pd-based membrane allows extraction of reactions products all along the reactor, and thus limits reverse reaction rate to the benefit of the direct one (shift effect). In such a process, it is possible to obtain a conversion rate of tritiated water up to nearly 90-95% and recover tritium under pure mixture of hydrogen isotopes. Based on former experimental studies, a phenomenological model was built to predict the efficiency of a fixed catalytic membrane reactor (CMR) design but this model does not allow optimizing the design of a CMR to reach fixed performance. However, the phenomenological model has given sufficient precise knowledge on the phenomena occurring in the CMR to allow the implementation of a CMR calculation module in the Prosimplus process simulation software environment. This paper describes the transfer step from phenomenological model to chemical process simulation tool and presents the methodology to be used to optimize the design of an industrial CMR under the Prosimplus software environment. The application of the proposed methodology for the design of a CMR for the Tokamak Exhaust Processing system of DEMO is given as an example.

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P2.186 DEMO Breeding Blanket Helium Cooled First Wall design investigation to cope high heat loads

In the framework of the European "HORIZON 2020" innovation and research program, the EUROfusion Consortium develops a design of a fusion power demonstrator (DEMO). One of the key components in the fusion reactor is the Breeding Blanket (BB) surrounding the plasma, ensuring tritium self-sufficiency, heat removal for conversion into electricity, and neutron shielding. CEA-Saclay, with the support of Wigner-RCP and Centrum výzkumu Řež, is in charge of the development of one of the four BB concepts investigated in Europe for DEMO: the Helium Cooled Lithium Lead BB.
The first component of the BB facing the plasma, the First Wall (FW), has until now been designed in order to respect the design criteria and temperature limit of the Eurofer structure for a maximum heat load extrapolated from ITER TBM, equal to 0.5 MW/m². New heat loads on the First Wall of DEMO BB have been assessed recently showing higher values on some poloidal location of the BB. This paper presents the investigation on the Helium FW design integrated to the BB (inlet helium

temperature at 300 ℃). Different designs have been studied from rectangular to circular channels and with different options for the tungsten armour surrounding the channels. The performance of the different concepts has been assessed with thermal and mechanical Finite Element Method numerical simulation based on simplified FW models and comparing results with the RCC-MRx code design rules to prevent failure during normal steady state condition and off normal condition in case of Loss Of Coolant Accident (LOCA) event. The results show that selected options with circular channel surrounded by tungsten could meet some of the requirements of plasma heat loads from design point of view. However, the concept is still in an early stage of development and open issues are discussed.

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P2.191 Optimal operation and pump regeneration plan of fusion fuel cycle based on the state-task network representation

The fuel cycle of the tritium plant has to safely handle the fuel gases including tritium and provide those gases to the fusion reactor. Given a required amount of tritium for fuelling scenarios considering ramp-up, flat-top, and ramp-down, a scheduling model is developed base on the state-task-network representation to provide the optimal operation plan for DT plasma operation including information on fueling rate, duration, and timing between each unit. The model aims to minimize tritium inventory including base inventory and working inventory inside the fuel cycle. The system scope of fuel cycle consists of Vacuum Roughing System, Tokamak Exhaust Process, Isotope Separation System, Storage Delivery System, and Fueling System. The model considers specific operation condition and tritium inventory limitation of each unit. The problem is formulated as a mixed integer nonlinear program (MINLP) model based on the state-task network representation with non-linear constraints. A case study of inductive operation mode is presented to illustrate the applicability of the proposed modeling and solution method. Presented case study is expected to provide many useful insights to determine the regeneration schedule of Vacuum Roughing System.

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P2.192 Design of Chinese Demo Blanket Concepts and R&D Progress of DFLL TBM

China has long been active in pushing forward the fusion energy development to the demonstration of electricity generation. As one of the most challenging components in DEMO, great efforts have been put on the development of breeder blanket and three blanket schemes were studied in China for fusion engineering test complementary with ITER (International Thermonuclear Experimental Reactor). In this paper, the main blanket concepts developed in China will be summarized including two leading schemes of Dual Functional Lead Lithium (DFLL) and Helium Cooled Ceramic Breeder (HCCB).

For ITER-TBM (Test Blanket Module), the Procurement Agreements of HCCB-TBM has now been confirmed and led by three institutes, i.e. SWIP (Southwest institute of Physics), INEST (Institute of Nuclear Energy Safety Technology) and CAEP (Chinese Academy of Engineering Physics), each playing a different role. The other candidate scheme, DFLL-TBM has also been continuously supported by CN-MOST (Ministry of Science and Technology) and will be tested in ITER under international cooperation.

The technical challenges to ITER-TBM and also the DEMO mainly focus on fusion material development and testing, breeder/coolant technology and experiment validation, effective tritium production/extraction to achieve self-sufficiency, reliability and safety etc., which are the nuclear technology basis of DEMO blanket. In this paper, the recent R&D progress on DFLL-TBM is presented, including the progress on structural material fabrication technologies and properties of CLAM steel,

PbLi/He coolant technology and safety issues, the RAMI analysis, small mockup neutronics experiments, and tritium behavior etc.. Based on the conceptual design and latest technical R&D progress, the entire test planning will be scheduled for DFLL concepts towards DEMO blanket.

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P2.194 Comparative study of thermal and microstructural properties of tungsten for the application to PFM

Tungsten is to be used as plasma facing material of divertor target for ITER. Though the present ITER specification for divertor target requires pure tungsten as a plasma facing materials, much research effort is being devoted to improve the material properties of tungsten for the application to severe conditions predicted in DEMO reactor. Among the material properties of tungsten, thermal conductivity is of primary importance in estimating the thermal response and stability of high heat flux component. While various methods are applied to improve the mechanical properties of tungsten, we should assess the thermal conduction of the materials and the resultant performance as a HHF component.
In this study, we investigated thermal and microstructural properties of pure and dispersion strengthened tungsten fabricated by Spark Plasma Sintering (SPS) method in comparison to the tungsten materials produced by different suppliers. We measured the thermal conductivity of the examined materials and compared them with chemical composition, density and microstructure of the samples. Thermal conductivity up to 900℃ was derived from thermal diffusivity and specific heat measurement by laser flash method. And chemical composition was analyzed through glow discharge mass spectroscopy. Thermal conductivity values show a close correlation with content of compositional element, on the other hand, Vickers hardness shows correlation with the average grain size of the samples. The sample produced by different fabrication process shows different density and thermal conductivity. While TiC-added tungsten fabricated by SPS method shows improved stability in microstructure against recrystallization after exposure to high heat flux, the reduction of thermal conductivity is analyzed in comparison to the amount of contained additive elements. In addition, we compared thermal conductance behavior through plasma facing component in high heat loading experiment with respect to measured thermal conductivity variation.

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P2.195 Development of the Dual-beam ion irradiation facility for FUsion materials (DiFU) at RBI Zagreb

The Dual-beam ion irradiation facility for FUsion materials (DiFU) is under development at the Ruđer Bošković Institute in Zagreb, Croatia, allowing irradiation of fusion-related samples by one or two ion beams. Two ion beams come to the DiFU chamber at an angle of 170 between them, from 6 MV HVE Tandem VDG and 1 MV HVE Tandetron accelerator. Ion beam handling and scanning systems enable fast electrostatic scanning of the beams over the sample at frequencies 256-3200 Hz in the horizontal and vertical axes, enabling the irradiation of areas from 5x5 to 30x30 mm2. The sample holder enables XYZ positioning of heated, cooled or room temperature samples. Ion fluxes are measured indirectly by insertion of two large Faraday cups in ion beams. Besides, the ion flux is monitored continuously by two sets of XY slits, which, in turn, define limits of the irradiation area on the sample. Sample temperature and conditions during irradiation are monitored by a set of thermocouples, an IR camera and a high-sensitive video-camera. The DiFU facility has been developed within EUROfusion Workpackage “Fusion materials”.

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P2.198 Grain scale constitutive modeling and simulation for reduced activation ferritic/martensitic steel

The constitutive behavior of Reduced Activation Ferritic/Martensitic (RAFM) steel for the potential blanket material of fusion reactor was modeled and implemented into a crystal plasticity finite element method (CPFEM). In the developed constitutive model, the plasticity was formulated by the slip system activation in the body centered cubic single crystal and the stress of polycrystal was homogenized based on the Taylor or full FE schemes. The model takes interactions among dislocations and precipitates into account, which is the main hardening mechanism of tempered martensite structure. Also, the damage accumulation by the decohesion of precipitate-matrix interface and subsequent growth was also phenomenologically included in the model. The proposed model was implemented into the crystal plasticity framework as a user material subroutine and validated by the comparison between simulated and measured flow stress curves for the RAFM steel. The microstructure used for the simulation was obtained from the EBSD data which showed the hierarchical microstructure consisting of laths, blocks and packets in the transformed martensite from prior austenite. Preliminary study showed that the developed constitutive model for the grain scale crystal plasticity FE simulation predicted the stress-strain response and fracture within reasonable accuracy when compared to the experimental data.

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P2.199 Overview of the Current Status of IFMIF-DONES Secondary Heat Removal System Design

The present paper summarizes the current status of the Secondary Heat Removal System (SHRS) of the IFMIF-DONES (International Fusion Material Irradiation Facility- Demo Oriented Neutron Source). As part of the Lithium systems (LS) in IFMIF-DONES, SHRS is the responsible sub-system for the removal of the heat which develops in the LS-Test Assembly (TA) during the Li - DT reaction. In this way, it has an important role in maintaining the stable lithium flow by providing controlled temperature at the inlet of TA.
In SHRS, a somewhat conventional hot oil system serves as an intermediate heat removal sub-system between the main liquid lithium loop and the water cooling system. The later serves as the final heat sink. During the preliminary design phase, the concept developed during the IFMIF-EVEDA phase served as a basis, but its applicability was studied with special regard to the effect of the difference in the transferred heat of IFMIF and IFMIF-DONES on the selection of the thermal layout, from point of view of cost, complexity and safety. The selected thermal layout consists of two intermediate oil loops with three heat exchangers and two different organic oils as heat transfer mediums. Following the definition of the thermal layout, the main units were selected, a conceptual pipeline layout was designed which corresponds to the current building configuration and a preliminary thermal-mechanical assessment was performed in order to assist the design of the supporting structures.

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P2.200 Irradiation of the copper-based high entropy alloys for nuclear fusion

Nuclear fusion is a promising way to fulfill the current and future energy needs in a cleaner way and many challenges must be overcome to be able to achieve a sustained nuclear fusion reaction. Tungsten with a high melting point, high sputtering threshold and low tritium inventory is the choice for the plasma facing material and CuCrZr alloy, with high conductivity and strength, for the heat sink material. The problem arises in the mismatch on the properties of both materials: the high ductile-brittle transition (DBTT) temperature for tungsten, while CuCrZr possesses a low service temperature. The thermal properties mismatch might lead to W-CuCrZr interface damages and lower time-life service for the materials used, and therefore, a thermal barrier layer is necessary. High entropy alloys have interesting properties such as low thermal diffusivity and form homogenous materials with simple structures, which put them as candidates for thermal barrier layers.
In this work, high entropy alloys were prepared with the composition CuxCrFeTiV (x = 5, 10, 20, 30 at.%) using mechanical alloying of mixes of elemental powders followed by spark plasma sintering at 1178 K and 65 MPa. One sintered CuCrFeTiV sample was irradiated at room temperature with 300 keV Ar+ ions to a fluence of 3×1020 at/m2. The samples were characterized by scanning electron microscopy coupled with energy dispersive X-ray spectroscopy and X-ray diffraction. Preliminary results showed the presence of heterogenous and multiphasic microstructures. The diffractograms of the as-sintered samples revealed two FCC structures the minor fraction with lattice parameter similar to Cu. After irradiation the CrCuFeTiV diffractogram revealed the coexistence the Cu-like FCC structure with a dominating BCC structure, implying/suggesting that the minor Cu-like FCC structure remained unchanged, while the original dominant FCC structure dissapeared, giving rise to a BCC structure.

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P2.202 Irradiation damage mechanism of tungsten studied by cold neutrons

Tungsten is the leading choice as plasma-facing material in fusion reactors. Its brittleness can be alleviated by alloying with few-% rhenium. The earliest nuclear application of the W-Re system was as thermocouples in experimental breeder reactors. A drop in efficiency of the thermocouples was noticed and attributed to radiation-induced precipitation. Many studies followed with varied conclusions on W-Re phases. The chi-phase precipitates formed under neutron irradiation are extremely brittle and can initiate points of cracking.

Under fusion neutron irradiation, pure W will convert into W-3%Re within 5 years due to transmutation reactions. This would change the material properties. Fission reactor studies have shown significant irradiation hardening. Needle-like precipitates in small void-free areas are often seen in spite of radiation-induced diffusion having occurred. This is in contrast to the Re induced ductilization. Accelerator irradiations, with W ions have also resulted in precipitate formation at large damage dpa. Density functional theory calculations using first principle studies have pointed at the W-Re dumbbell three dimensional movement as being responsible for the Re precipitation far below the solubility limit.

In order to separate effects caused by transmutation (alloy formation) and by collisional damage, experiments with low-energy ("cold") neutrons can be useful and are often used to investigate i.e. biological samples due to their non-destructive probing nature. Cold neutrons don't incite displacement damage in tungsten, but only induce Re through transmutation. An irradiation using cold neutrons to induce Re precipitation without accompanying vacancy/interstitial creation leads to a deeper understanding of the W-Re system in a neutron environment. Isotopically pure 186W samples are exposed to cold neutrons in several locations at FRM-II, Garching. Subsequent atom probe tomography will evaluate the distribution of Re atoms and precipitates from transmutation. Sample preparation, activation, dose, recoil distribution simulations and pre-irradiation atom probe tomography study are discussed in this contribution.

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P2.203 Conceptual design of STUMM module for characterization of neutron and gamma radiation fields during commissioning phase of IFMIF DONES

IFMIF-DONES (International Fusion Materials Irradiation Facility — DEMO-Oriented Neutron Source) will be built as a powerful neutron source to test suitable materials planned for the construction of future tokamaks like DEMO (Demonstration Fusion Power Plant). In the commissioning phase of IFMIF-DONES it is foreseen that a Start-Up Monitoring Module (STUMM) will be used for the characterization of neutron and gamma radiation fields.
The STUMM will be positioned inside the Test Cell just behind the DONES neutron source. It will be at the same position as the High Flux Test Module (HFTM) that planned for irradiation of samples. As a consequence, STUMM will be operated in extremely harsh radiation and high temperature conditions.
The main mission of STUMM is to characterize the neutron source (determine the energy and space distribution of neutron and gamma fluxes), to characterize the radiation fields at the location of the HFTM and to verify the results of neutronic modelling of those distributions. To fulfill its mission STUMM will be composed of selected detection systems and sensors characterized by a long radiation resistance and relatively small radiation sensitivity. At this stage of STUMM conceptual design following detection systems are foreseen: Rabbit System, N-16 system, gamma thermometers, SPND detectors, micro-fission chambers, thermocouples and strain gauges.
The conceptual design of STUMM is prepared as part of the Eurofusion Early Neutron Source work package (WPENS) in collaboration between engineers and physicists from the Institute of Nuclear Physics Polish Academy of Sciences (IFJ PAN) and the National Centre for Nuclear Research (NCBJ).

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P2.204 Results of 16 MeV proton irradiation on tungsten for fusion relevant damage

Fusion first wall materials are required to withstand large neutron damage during their lifetime. This damage comprises of knock on induced displacement damage and a material composition change through transmutation reactions. Protons of up to 5 MeV and heavy ions from accelerators are capable of reproducing displacement damage similar to that from neutrons in a fusion reactor. As the accelerator produced protons register significantly higher damage rates than fission reactors, they are used to simulate the fusion neutron damage. Due to the low proton energy, the transmutation reactions are not reproduced and no change in material composition is observed. However, raising the proton energy above reaction thresholds, such as 16 MeV on tungsten, could introduce transmutation effects into the field of study. Higher energy simultaneously increases the damage depths to macroscopic levels of above 300 μm.
By varying the energy of the protons employed, a parameter study can be adapted at accelerators to understand the development of material changes through irradiation. In view of the above, a 16 MeV proton irradiation was undertaken, additionally serving the purpose of testing sample geometry, sample holder stability, heat removal, vacuum tightness and sample transport. All studies were simulated using MCNP6.1, FISPACT-II and SPECTRA-PKA calculations.
A macroscopic sample design was conceived to accommodate multiple tests within a single irradiation exposure. The sample sizes incorporated correspond to the recommended dimensions as prescribed in fission reactor studies. A real time temperature measurement system is incorporated within the sample holder. To obtain dose rate, activity levels and heat generation during and post

irradiation, the entire setup was modelled in MCNP6.1.
The sample setup was tested with 16 MeV protons and analysed for mechanical property changes in the hot cells. These initial steps, including design of samples, holder and corresponding test results, will be presented in the talk.

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P2.206 Precipitation phenomena during corrosion testing in the forced-convection Pb-15.7Li loop PICOLO

Several blanket concepts (e.g., HCLL, WCLL, DCLL) are based on the application of the liquid breeder Pb-15.7Li, which is in direct contact with the structural components. Compatibility testing has shown that the structural materials (e.g., Eurofer) always suffer from corrosion attack which mainly depends on the operation temperature and flow velocity of the liquid breeder. The governing mechanism can be attributed to dissolution of the steel by the liquid breeder. At high flow velocities (v = 0.1 m/s), high dissolution rates of ca. 200 µm/year were evaluated at 823 K. This corresponds to an amount of ca. 1.6 kg/m² of removed steel constituents. A fusion device with blanket modules or also a corrosion testing loop, e.g., the PICOLO loop of KIT, are non-isothermal systems. The components dissolved at higher temperatures are transported with the breeder flow. At sections with a lower temperature oversaturation of the breeder will occur. Effects such as deposition, precipitate formation and transport of corrosion products will take place. Corrosion testing in PICOLO showed that such particles are formed and that they can seriously affect the safe and reliable operation of Pb-15.7Li systems up to blocking of components. Thus, during maintenance work tubing sections and components beyond the magnetic trap, which is installed and designated in PICOLO to collect formed particles, were removed for analyzing the issue of occurring precipitates. Particles were analyzed by SEM/EDX and metallographic techniques concerning composition, the areas of deposition, shape and size. The comparison of corrosion/deposition modelling with areas of detected deposition indicated that a transport of particles is present. The observed effects will also be discussed with respect to the development of new purification components and further testing needs towards a reliable operation of blanket systems.

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P2.207 Properties of boron carbide ceramics made by various methods for use in ITER

One of the functions of ITER diagnostic port-plugs is neutron protection of the equipment installed in the port, as well as reducing the radiation background in the area of reactor elements requiring access for maintenance personnel. Engineering restrictions on the full weight of the port plug and the amount of water in the reactor do not allow the use traditional iron-water protection. Boron carbide is an alternative protection material. Due to the low atomic weight and high absorption cross-section of thermal neutrons, boron carbide can serve as an effective flux attenuator for both fast and thermal neutrons.
Properties of ceramic based on boron carbide essentially depend on the technology of its manufacture. In this regard, for decision-making on the possible use of a certain type of ceramics in ITER it is necessary to conduct measurements of the elemental composition and physical properties of ceramics thermal conductivity and outgassing rate in a vacuum.
Within the framework of the work, research was carried out on ceramics based on boron carbide manufactured at Russian enterprises. Samples of ceramics were submitted by Virial (St. Petersburg), NEVZ-Ceramics (Novosibirsk), Beeforce (Tver), Izomed (Moscow). Hot-pressed, reaction-bonded and sintered boron carbide ceramics as well as the initial powder were studied.
The elemental composition of the samples and the impurity content were investigated by scanning

electron microscope with energy dispersion attachment and by the amount of x-ray radiation attenuation in the samples. In addition, measurements were made of the outgassing rate into the vacuum and mass spectrum allocated to the residual gas and contact conduction thermal transitions ceramic-on-ceramic and ceramic-stainless steel.
It was found that in hot-pressed and sintered boron carbide impurities are not more than 1%, in reactive boron carbide the content of impurities can reach 10-15% (mainly silicon or oxygen in different types of ceramics).

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P2.212 Effects of specimen thickness on high-temperature tensile and creep properties of F82H reduced-activation ferritic/martensitic steel

A reduced-activation ferritic steel, F82H steel, is the primary candidate structural material for fusion blanket. Neutron irradiation properties are estimated by using miniature specimens. Since the thickness of the gauge section of the miniature tensile specimens and of wall thickness for creep tubes is less than 1 mm, deformation volume is much smaller than that of standard size specimens. The present study seeks the effect of thickness of the miniature specimens on tensile tests and creep properties. In addition, effects of specimen size are discussed based on comparisons with standard specimen data.
SSJ type specimens with a gauge length of 5 mm and a gauge width of 1.2 mm were machined from a 15 mm-thick plate of F82H-IEA heat. The gauge thickness was varied from 0.14 mm to 1.2 mm. Tensile tests at room temperature (RT) and 650℃ were conducted. The initial strain rate for the tensile tests was 6.7 x 10^-4. Creep tests were performed under 75 to 120 MPa at 650℃. Some of the tests were terminated at 100 h to measure the creep strain in direct with digital microscope.
Effects of specimen thickness on yield strength, ultimate tensile strength and uniform elongation were not significant in the tensile tests at RT and 650℃, while total elongation was obviously decreased with decreasing specimen thickness. Less and asymmetric necking was observed after fracture for the thinner specimens, and indicated much localized deformation, while extensive and symmetric necking occurred for the thicker specimens. The less necking deformation is probably caused by less activation of shear slip bands, and lead to the degradation of total elongation. Increase in creep strain was observed for the thinnest specimen after 100 h test under 120 MPa at 650℃. Effect of the surface on creep deformation will be discussed.

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P2.211 Fatigue characterization and modeling of CLAM steel under multi-axial non-proportional cycle loading

Due to the low activation and excellent neutron irradiation resistance, the Reduced Activation Ferritic/Martensitic (RAFM) steel has been considered as the primary candidate structural material for the blanket of the first fusion reactor plant. The rigorous serving condition of a fusion reactor lead to to multiaxial stress-strain condition of the blanket. With the multiaxial cyclic loading, the RAFM steel possesses additional hardening effect which will deteriorate the material performance.
In this work, multiaxial fatigue tests were conducted on China Low Activation Martensitic (CLAM) steel under proportional and non-proportional loading. Additional hardening was observed under non-proportional loading condition. Under torsion loading, large numbers of early micro cracks emanated from the sample's surface, and a few micro cracks propagated into very long longitudinal cracks. In biaxial tests, cracks tended to propagate into the gauge reducing the cross-section area. A strain-energy-based critical plane fatigue lifetime prediction model has been employed for life prediction of the material to account the effects of the additional hardening effect caused by the

multiaxial load. The simulated evolution curves of the multiaxial damage and the predicted fatigue lives are in good agreement with the experimental results of CLAM steel.

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P2.222 Analysis and possible reduction of fusion plant construction costs

In fusion plants the overnight cost (capital cost minus financing) is expected to be the main contribution to the cost of electricity. The overnight costs (euro/kWe) for ITER and DEMO are several times higher than the ones of present commercial sources (wind, photovoltaic, fission, coal, gas, ...). It is shown that cost reduction from high learning rates in future commercial plants should not be taken for granted, because of the size and complexity of fusion plants. Therefore, assuming the remaining engineering issues are solved, the economic aspect is highly relevant already at the level of a DEMO plant. In order to identify the main sources of cost in fusion plants, two type of analysis have been considered: in the first approach the costs are subdivided between plant core and auxiliary equipment. In the second one the focus is on the specific material input for the components (quantified in tons of steel per kWe) and the associated specific cost (in euro/ton), which can vary greatly depending not only on type of material but also on the specifications, regulations and quantities (for example competition among suppliers). Comparisons are made with commercial power plants and in particular with fission and coal plants, which share with fusion plants some common features (large capital costs, large components, extensive on-site construction). From both analysis (core/auxiliary equipment and specific material input/specific costs) fusion plants looks clearly more expensive than fission and other commercial plants, mainly because of the low power density and complexity. Nevertheless, the analysis helps to identify some possible paths leading to a reduction in the construction costs, bringing fusion power plant on the way to compete with traditional electricity sources.

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P2.037 Automatic identification of the plasma equilibrium operating space in tokamaks

Identifying the plasma equilibrium operating space in terms of e.g. plasma current Ip, internal inductance li(3) and magnetic flux state ψ is a central task in the design of future tokamaks. The operating space is typically limited (for a given plasma shape) by constraints on the Poloidal Field (PF) system such as maximal allowable currents, fields and forces in PF coils. The typical tool to identify this operating space is an inverse Free-Boundary Equilibrium (FBE) solver, which is used to find a set of PF coils currents that minimizes a cost function comprising two terms: the first one, which we call here the objective, quantifies the distance between the actual and desired plasma shape and the second one is a regularization term, typically a weighted sum of the squares of the PF coils currents. This formulation does not however directly take into account the above-mentioned PF coils limits. Identifying the operating space in these conditions is time-consuming (usually several days) and needs heavy human intervention to tune the weights in the cost function.
Here, we replace the regularization term in the cost function by new terms which penalize the violation of the limits in PF coils currents, fields and forces. This way we can obtain the operating space by scanning parameters (e.g. Ip, li(3), Ψ), running an inverse FBE problem for each point in the scan. The operating space is then defined by an iso-value contour of the shape deviation metric (this contour being entirely located within all PF limits).
As an example, we applied our method to the ITER 15 and 17 MA scenarios and found, within a few hours of simulation and with little human intervention, results similar to previous studies.

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P2.039 Design of MITICA control and interlock systems

The construction of MITICA, the full-size prototype of the ITER Heating Neutral Beam Injectors (HNBs) is in progress at the Neutral Beam Test Facility (NBTF) located in Padova, Italy.
The design of the central control (CODAS) and interlock (CIS) systems is progressing taking into account the requirements coming from the MITICA plant units, in terms of number and type of interface signals, data sampling frequency values, operating states, protection actions and required reaction times and reliability. As the HNBs will be ITER components, additional requirements derive from the ITER plant control design handbook that specifies the design criteria and technologies to be applied to ITER plant systems.
In MITICA a distinction must be made between plant units that will be part of the HNB, such as the power supply systems, and plant units that are auxiliaries, such as the gas and vacuum system. In the NBTF these components shall interface with MITICA CODAS and CIS, whereas in ITER they will be autonomous plant systems providing services to the HNB. The distinction reported above suggests defining the architecture of MITICA Instrumentation and Control (I&C) as split into the following functional blocks:
- HNB plant system – including components that will be part of the HNB;
- Auxiliary plant system – including all operational plant units that need control in the NBTF but are implemented outside the ITER HNB;
- Diagnostics plant system – including the diagnostic instrumentation
- Central I&C (CODAS and interlock)
- Communication infrastructure.

The paper will present the requirements for MITICA CODAS and CIS and will derive the detailed system architecture. The design of CODAS and CIS will be discussed with reference to the lesson learned in the implementation of SPIDER CODAS and CIS (the NBTF ion source test bed) and to the technologies suitable for the implementation.

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P2.225 Assessment on radioactivity source term release and public impact for CFETR

CFETR is now in engineering design phase (EDP) and will hopefully complete this phase around 2020. It is essential to evaluate the public impact due to the radioactivity release in this stage. In this work, the radioactive inventory and property of source terms were evaluated, discussed and compared with ITER. Then, under normal operation radioactivity release limit was researched comprehensively considering the Chinese regulation for public individual dose limit and exhausting from nuclear power plant. And for accident condition, potential risk of the radioactivity release was also evaluated preliminarily. The public and environmental impact were estimated by atmosphere dispersion modelling and migration along food chains. Countermeasures were also discussed based on the objective that to eliminate emergency evacuation for fusion power plant. As a conclusion from this assessment, the safety objectives and requirements were proposed for CFETR to ensure nuclear safety.

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P2.216 Effect of Hydrogen on Corrosion Properties of Reduced Activation Ferritic/Martensitic Steel F82H

Reduced activation ferritic/martensitic (RAFM) steel, e.g., F82H, is the leading candidate structural material for fusion blanket. Of many blanket concepts, the water-cooled ceramic breeder blanket is an attractive concept because of its compactness and its compatibility with the technologies in conventional light water reactor. For tritium breeding, it is necessary to manage the corrosion of F82H by controlling not chemicals but hydrogen and oxygen in the cooling water. Understanding the effects of oxygen and hydrogen on corrosion property was therefore required for the blanket design. This work aims to investigate the corrosion behavior of F82H in high-temperature water with hydrogen added.
The material used in this study was Japanese RAFM, F82H. The flow-accelerated corrosion (FAC) test was performed using a rotating disk specimen (O.D. = 275 mm, I.D. = 130 mm, thickness = 3 mm). The circumferential velocity on the specimen edge was 5 m/s. The FAC tests were conducted at 543 K under the pressure of 6.3 MPa. The corrosion fatigue test was also performed under 0.6 % of the total strain range and 0.0004 %/sec of the strain rate with a triangle waveform at 573 K under the pressure of 13 MPa. The dissolved oxygen (DO) and dissolved hydrogen (DH) concentrations in the water were <5 ppb and 3.5 ppm, respectively, for both experiments.
The weight of rotating disk specimen in the high-temperature hydrogen-added water was decreased with increasing time as previously reported in [1] (573K, DO 20ppb). It is specurated that no considerable change of FAC was expected by hydrogen addition. In contrast, it was found that the number of cycles to fracture was decreased in the high-temperature water compared with the fatigue test at room temperature in air.

[1] M. Nakajima,et. al. Journal of Plasma and Fusion Research SERIES, 11 (2015) 69-72.

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P2.041 DCS Satellite: Enhanced Plant System Integration on ASDEX Upgrade

The integration of plant systems and the discharge control system (DCS) will become more important for future fusion experiments.
Whereas nowadays experiments can mostly operate with a separation of diagnostics and control, a higher level of integration of different systems is neccessary in the future.
In ITER comprehensive information about the plasma system state will be required at all times.
ASDEX Upgrade is extending the integration of diagnostics, actuators and evaluation codes in a unified concept, the DCS satellite.
Within the DCS satellite, the means to share information between different systems is provided in a unified way by the DCS framework.
This allows automated configuration and identification of incompatibilities in the current plant system setup prior to the discharge execution.
During a pulse the availability of information provided by this framework enables the generation of a comprehensive plasma state which then can be used for control.

The framework is developed in C++ using Linux with real time kernel as an operating system.
This allows the best compromise between ease of use and flexibility in the choice for the diagnostic setup.
The DCS satellite concept benefits from the ability of the DCS framework to incorporate different data types and rates, which is important since not all DCS satellites can deliver their data at the same rate.

So far this concept has successfully been demonstrated for National Instruments NI-RIO based FPGA diagnostics, video real time diagnostics (VRT) using CameraLink cameras, diagnostics using IPPs inhouse standard SIO2 as well as diagnostic codes such as the equilibrium reconstruction code JANET++.

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P2.219 Tritium and Dust Source Term Inventory Evaluation Issues in the European DEMO reactor concepts

Fusion reactors represent a future evolution of the nuclear technology improving the world-wide energy portfolio. The experimental fusion reactor under construction (ITER) and the planned industrial fusion reactors (DEMO) are large and complex facilities. For their operation it is necessary to ensure safety and solve several technical issues. The limitation of the radiological and mobilizable source terms inventory, tritium and radioactive dust is one of them.
The source term inventories shall be assumed in the establishment of the operational and safety requirements for DEMO, as well as performing the safety analyses for the commercial fusion devices. In the last few years a methodology for the evaluation of tritium and dust source term inventory has been proposed in the framework of the EUROFusion project. The basis of the methodology is a semi-empirical approach to scale the radioactive inventories limits implemented in ITER. The amounts derived according to this methodology will supply a guidance for the safe operation of the future fusion reactors. The development of this methodology has to be completed and refined because of the lack of a validation versus real experimental data and rules for the extension to different scenarios.
The aim of this work is the validation of the developed methodology for the evaluation of tritium and dust source term inventory regarding appropriate experimental facilities (such as JET and ASDEX).

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P2.224 Mesh based Variance Reduction Technique in Shielding Calculations of the Stellarator Power Reactor HELIAS

The Helical-Axis Advanced Stellarator (HELIAS) is the leading stellarator concept in Europe and developed at the Max-Planck-Institute for Plasma Physics (IPP). Based on the 5-field-period symmetry, the HELIAS-5B engineering design study emerged which aims at a stellarator power reactor designed for 3000 MW fusion power.
The stellarator confines hot plasma only by external superconducting field coils, which are very sensitive for the neutron flux, leading to a complex 3D topology of the magnetic configuration. These coils define the shape of the device and limit the space available outside the plasma chamber for the vacuum vessel and the in-vessel components required for breeding, shielding and heat removal. It is thus necessary to identify the locations where minimum space for the shielding is available and check there the radiation loads to the magnetic field coils in order to proof a sufficient shielding capability.
This requires a suitable computational approach to simulate the generation of source neutrons in the plasma chamber and the neutron transport through the complex HELIAS geometry. The approach is based on the Monte Carlo (MC) particle transport technique applied with the Direct Accelerated Geometry Monte Carlo (DAGMC) method. The particle transport to regions far away from the plasma source typically results in higher statistical uncertainties when using the standard analogue MC particle transport simulation technique. The statistics can be greatly improved by running non-analogue calculations with the application of suitable variance reduction (VR) techniques.
The paper presents results of the shielding calculations performed for HELIAS and discusses the application of the mesh based weight window VR technique to achieve statistically reliable tally results in the region of the non-planar shaped field coils. The calculated nuclear responses are compared to the design values specified as limits for the radiation loads to the superconducting toroidal field coils of the DEMO tokamak.

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P2.229 Methodology of probabilistic risk assessment for tokamak-type fusion reactors

The world's largest tokamak fusion device-ITER is under construction in Cadarache, France, and first plasma will be officially identified in 2025. Building on the work of ITER, various countries are planning the steps needed for the fusion demonstration reactor (DEMO), and many conceptual designs for the fusion power plants (FPP) have been developed, such as PPCS of the European Union, ARIES of the United State and FDS of China. Importantly, the societal risk indicator of nuclear accidents has not been well understood and assessed for the tokamak-type FPP. Probabilistic risk assessment (PRA), which has long been applied in fission nuclear reactor development, would be a promising way for this object.
In this contribution, a typical deuteron-tritium tokamak FPP was selected for this study. Initiating events under the full power operation conditions were identified after elaborate analysis of mobilizable radioactive materials, energy sources and confinement barriers. Possible accidents evolutions for these initiating events were expanded by event tree method, and the accidents sequences were categorized by their environmental release routes. Representative initiating events and accidents sequences for all release routes were identified based on the possible consequences, and the frequencies of accident sequences due to each initiating event were thus calculated. Doses of representative accidents were assessed according to the average (best estimate) meteorology defined by DOE, and a complementary cumulative distribution function (CCDF)-based risk curve was obtained reflecting the societal risk, which describes the frequencies of exceeding given doses summed over the contributions to risk from the entire spectrum of accident sequences. It is illustrated that, for the tokamak-type FPP, PRA method could also be an effective risk assessment method, and provide insights and suggestions in its design and development.

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P2.230 Thermal-hydraulic modeling and analyses of the water-cooled EU DEMO using RELAP5 system code

The current conceptual design of the Primary Heat Transfer System (PHTS) of the water-cooled EU DEMO foresees two independent cooling circuits, the breeding zone PHTS and the first wall PHTS. During the pulse (120 minutes) the first delivers thermal power to the turbine, the latter delivers thermal power to the Intermediate Heat Transfer System (IHTS) equipped by an Energy Storage System (ESS). The IHTS delivers partially thermal power to the turbine in pulse so that to accumulate a suitable amount of energy in ESS to operate the turbine during the dwell time (10 minutes) at almost constant load, despite the EU DEMO pulsation of the generated thermal power.
A dynamic model of the primary systems of water-cooled EU DEMO is developed using RELAP5 system code, capable to perform steady state and transient simulations. The model includes the primary and secondary side of the breeding zone; first wall primary system and the intermediate and energy storage systems are included too. A detailed nodalization is carried out for in-vessel components (i.e. breeding blanket), where higher accuracy is required. The ex-vessel components are modeled maintaining real geometries (i.e. main collectors, hot and cold legs) or adopting equivalent components (i.e. heat exchangers and steam generators tubes). Heat structures are used to model the heat transfer between primary and secondary side. Preliminary assessments of the nodalization have been carried out, in particular checking pressure drops along the systems and heat exchanger performances. The system analyses are performed to investigate postulated operation of EU DEMO power plant (i.e. pulse to dwell transient) and to evaluate advantages and key issues of the adopted configuration of the primary system and power conversion system.

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P2.044 Hazard function exploration of tokamak tearing mode stability boundaries

It is possible to decompose an event prediction problem into a hazard function (event rate model) and a phase space trajectory (dynamical system evolution). In contrast to typical event prediction approaches (such as those attempted in present tokamak disruption systems) the hazard function has two significant advantages. First, it has a time localized and quantitative interpretation (events per time) which can block spurious future-to-past statistical associations. Second, event rate models can be used to generate event predictions for arbitrary look-ahead horizons conditioned on future controls with respect to a dynamical systems model. In principle, this property allows a formalized approach to control design for high tokamak performance with dynamic avoidance of operation near the (hazardous) parameter regions where the event risk is significant. To be able to exploit these advantages, we are first required to develop the necessary tools to effectively learn (correct) hazard function models from data.
Previous results with hazard function models have been obtained from DIII-D at reduced scale (about 200 shots) for particular ITER-similar baseline scenario plasmas for tearing mode onset events. Extending this, we report the first hazard function analysis at large scale applied to tokamak data for tearing mode onset characterization. Newly developed (and massively parallelized) shot-database searching and feature extraction tool-chains at the DIII-D tokamak are exploited. Multiple methods for understanding and visualizing the properties of the estimated hazard function(s) are applied, including partial dependence plots, and verification of statistical calibration, which is a prerequisite for plausible application to control designs.
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P2.233 First considerations on the Balance of Plant for a HELIAS power plant

With the advance of fusion research both in physics and engineering and the start of the conceptual design development of respective fusion plants it becomes important to study how fusion will interact with the energy system. However, in order to study the interface between a fusion plant and the energy grid, the details of the Balance of Plant (BoP) must be known and modeled.
Recently, first studies of the BoP for a pulsed tokamak DEMO have been started. Following the DEMO approach, first studies for a stellarator BoP will be presented in this work. In contrast to pulsed tokamaks, stellarators are by design intrinsically capable of steady-state operation. Consequently, stellarators are attractive candidates for a direct-coupled-plant design (DCP) which directly links the primary heat transfer system (PHTS) and the power conversion system (PCS). However, fusion needs to be compatible with the future energy system (Energy System Integration-2050+) that will face significant challenges to balance loads due to the high share of fluctuating sources from massive variable renewable energy sources (VRES). Consequently, the need for sources increases which can be tuned in a flexible way to react on a fast time scale to the needs of the energy system. Therefore, a BoP with an intermediate heat and storage system (IHTS) using an integrated thermal energy storage system (ESS) seems attractive to cope with such demands. However, detailed modeling is required to assess how an ESS can be well coupled with the PHTS. Possible options for the storage could be a solid carrier medium or liquid metals. In this respect, an indirect-coupled-plant design (ICP) seems unnecessary for a stellarator, but remains to be studied.
In this work the BoP with an IHTS is modeled for the HELIAS power plant concept using the industrial software EBSILON® (STEAG).

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P2.234 Validation of SIMMER-III code for in-box LOCA of WCLL BB based on Test D1.1 of LIFUS5/Mod3 facility

One of the four breeding blanket candidates as options for European DEMO nuclear fusion reactor is the Water Coolant Lithium Lead Breeding Blanket (WCLL BB). The WCLL in-box LOCA (Loss of Coolant Accident) is a major safety concern of this component, therefore transient behavior will be investigated to support the design, to evaluate the consequences and to adopt mitigating countermeasures. This requires the availability of a predictive tool code capable to simulate the safety related phenomena and parameters occurring during the transient. To fulfill this objective, at the beginning step, SIMMER-III code was improved implementing the chemical reaction between PbLi and water. Then, SIMMER-III Verification and Validation (V&V) procedures have been established and conducted to obtain a qualified code for deterministic safety analysis. The verification activity was successfully completed, while the validation activity requires further efforts according with the R&D plan set up in the framework of the EUROfusion Project. In view of this, an experimental campaign and a test matrix has been designed in LIFUS5/Mod3 facility.
The experimental data of the Test D1.1 which are obtained from the mentioned facility are used for the validation SIMMER-III according with a standard procedure. This is based on a three-steps involving the qualitative and quantitative evaluations of code results. The qualitative accuracy evaluation is performed through a systematic comparison between experimental and calculated time trends based on the engineering analysis, the resulting sequence of main events, the identification of phenomenological windows and aspects relevant to safety. Finally, the accuracy of the code prediction is evaluated from quantitative point of view by means of selected figures of merit.

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P2.236 Biomass gasification with high temperature heat and economic assessment of Fusion-Biomass Hybrid System

This paper aims to make a further investigation on economic analysis of fusion-biomass hybrid model based on previously reported concept. This system postulates to gasify cellulose and lignin which are compositions of biomass derived from agricultural and forestry sectors to produce synthetic gas for artificial diesel and hydrogen production. Gasification process (C6H10O5 + H2O ⟶ 6H2+ 6CO) requires over 700oC of high temperature that blanket concept with LiPb and SiC technology such as DCLL is applicable to deliver high thermal heat for endothermic gasification reaction. This technical extension enables to produce synthetic gas which converts into either artificial diesel by Fischer-Tropsch reaction (2H2+ CO ⟶ -CH2-+ H2O) or hydrogen by water-gas shift reaction (CO + H2O ⟷ H2+ CO2).
While many of the previous application processes of biomass applied partial oxidation, this gasification reaction is endothermic when the reactant is isolated from oxygen, and can be regarded as fusion energy conversion. A 30-year lifetime cost of fusion-biomass hybrid plant is analyzed and compared with fusion electricity that typically converted at 33% of thermal efficiency. Economic analysis method of Discounted Payback Period (DPBP) and Internal Rate of Return (IRR) describes the profitability of fusion-biomass hybrid plant with low feedstock cost from waste biomass suggests attractiveness of fusion biomass hybrid.
Thermogravimetric analysis (TGA) and differential thermal analysis (DTA) of cellulose and lignin to assess accurate amount of heat absorption, as the function of temperature will also be reported. In general, gasification at lower temperature yields more complicated chemical products than simple C1 products with smaller absorbing heat. Since reduction in CO2 emission is expected by replacing fossil, fusion-biomass hybrid plant for clean fuel production can be attractive to deploy, because the role of liquid and gaseous fuels in transportation and industrial sectors can provide additional and potentially larger market chance for fusion.

P2 / 808

P2.237 Experimental measurements of pressure temperature and dust velocities: comparisons with a multiphase numerical model

The production of dust inside the nuclear fusion power plants is one of the safety issues of this technology. Dust is generated because of plasma-material interactions and deposits in the bottom regions of the TOKAMAK. In case of a Loss Of Vacuum Accident (LOVA), the dust may be resuspended, threatening the functioning and the safety of these reactors. A deep study of this phenomenon is required to develop counter measurements and to improve the safety of this promising way to produce energy. The authors have studied the fluid dynamics of these accidents with a scaled experiment, called STARDUST-Upgrade. Optical techniques have been implemented to measure dust resuspension and diffusion properties, such as velocity vectors and resuspension-rate. In this work, the authors performed a numerical multiphase fluid dynamics simulation with a commercial code, using a Euler-Euler approach, a Schiller-Naumann resistance model, and a k-ε turbulence model. The dust used is carbon (graphite) dust, that has been placed close to the inlet valve in both cases (numerical and experimental). The numerical results are analysed and compared with the experimental ones and the main agreements and differences are highlighted. The results show good accordance about the velocity vectors of dust, while the resuspension rate is overestimated in the numerical case because of the absence of adhesion and cohesion forces between dust particles and walls. This analysis is the starting point for the evolution and completion of a numerical model suitable for dust resuspension in case of LOVAs.

P2 / 619

P2.047 JET FIR Interferometer laser operation and interlock system upgrade to an open automation system

We report on the selection, implementation and successful demonstration of a new automated laser control system for JET's Far Infrared Interferometer, a diagnostic essential for machine protection. The new control system allows all laser subsystems and sensors to be interlocked and operated remotely in a precise and preprogramed manner, functionalities that are essential for reliable operation with the additional access restrictions foreseen in the upcoming D-T operation of JET. Our selection process gave priority to known technologies that conform to industry standards and have been proven within the JET operational environment. Weighting was also given to the accessibility of the software interface and the scalability of the technology.
The new laser control system was installed and tested at the end of 2017. It comprises an Industrial PC, running Windows Embedded Standard 7 and a real-time Programmable Logic Controller (PLC) using the proprietary protocol TwinCAT developed by Beckhoff. This technology is extremely compact (DIN rail mountable), scalable, modular and does not require additional wiring between modules. A particularly noteworthy feature is the ability to program the logic via higher level languages, in addition to standard PLC programming techniques. In this example, the main code for control and monitoring of the system was developed in Python using Finite State Machine algorithms.
This technology has application beyond laser control in a wide variety of application areas where reliable and robust remote operation and easy maintenance/service is a priority. Indeed, similar systems have been reliably used in the JET Plant Essential Monitoring System since 2015.

This work has been carried out within the framework of the Contract for the Operation of the JET Facilities and has received funding from the European Union's Horizon 2020 research and innovation programme. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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P2.049 A comparative study of different deconvolution methods used for reconstruction of neutron spectrum

Portable neutron generators (NGs) are widely used in many applications e.g. medicine, materials analysis and plasma diagnostics calibrations. The NGs based on DT reaction provide a controllable 14-MeV neutron emission of high-intensity flux. The knowledge of the total neutron yield is important, and in some of the applications energy distribution of produced particles are also essential. Neutron measurement techniques like activation method provide the information about total neutron emission. The reconstruction of the neutron spectrum is more difficult and requires sophisticated deconvolution methods which are unstable and should be carefully chosen for the specified case. The measurements are characterized by nonzero uncertainty and usually, the number of data is bounded thus causing the problem to be extremely ill-conditioned.
The activation foils were irradiated in the flux of 14-MeV neutrons emitted by NG working in continuous mode. The results of the activation measurement have been used for the reconstruction of emitted spectrum by different deconvolution methods. Minimum Fisher Regularization, Maximum Entropy Method and its connection with Conjugate Gradients Method, Maximum Likelihood Expectation Method and most researched Tikhonov regularization have been implemented in the Python programming language algorithms which were used for the reconstruction of the spectrum emitted by NG. The Codes have been tested with simulation data before the main analysis of the measurement. The results of the reconstruction of own-made codes and UMG Package programs have been compared. The obtained neutron spectra are not identical.
The deconvolution is a very complex problem, requiring experience in inverse problems and understanding the theory of the unfolding methods to correctly evaluate the solution. Discovering the most reliable solution also requires a good knowledge of the phenomena in the object under study. The described deconvolution methods can be implemented to the reconstruction of the neutron spectra from fusion devices like tokamaks or stellarators.

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P2.208 Microstructure and mechanical property mapping of CuCrZr with complex and non-uniform thermal history

The divertor component is subject to some of the most extreme loading conditions in a fusion environment and the safety design window is relatively narrow. Variation in mechanical properties of the same material throughout a component is common place in practice and can be caused by both local effects of processing and joining, and local variations in thermal history, neutron flux and other environmental conditions. By their nature, divertor geometries exhibit high gradients in temperature and stress due to high heat loading and geometric discontinuities, which is readily identified by conventional engineering analysis tools such as finite element modelling. The complex conditions imposed on materials can result in spatially heterogeneous properties which are difficult to predict and remain unaccounted for in conventional structural integrity assessments.
In this work, we present the use of multiple techniques including small angle neutron scattering (SANS), Transmission Electron Microscopy (TEM), Nanoindentation (NI) and tensile testing, in order to relate variation in materials microstructure with tensile properties for CuCrZr alloys with controlled thermal history. Then we apply SANS and NI to determine the microstructural state typical of a divertor mock-up assembly, which has undergone complex heating and cooling cycles. Material property mapping is used to measure the local variation in properties throughout a component geometry, which is used to infer a variation in local stress-strain properties. It is proposed that the use of measured spatially heterogeneous material properties by these mapping techniques may provide engineers with useful insights for the design and failure analysis of components.

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P2.209 Measurement of thermal conductivity of Li2TiO3 pebble bed by laser flash method

The functional materials of solid-type breeding blanket concepts for fusion reactor are used in a pebble bed form. In order to verify the performance and safety of the breeding blanket, the thermal conductivity of pebble bed is required. The hot wire, hot disk, guarded hot plate and laser flash method are considered as measurement technique for the thermal conductivity of pebble bed. This study aims at the provision of thermal conductivity of Li2TiO3 pebble bed for the breeding blanket. The laser flash method has been chosen due to its advantages compared with other methods, such as high accuracy and repeatability, easy sample preparation, and absolute measurement technique. In this method, the thermal diffusivity is directly measured by the laser flash system, and then the effective thermal conductivity is obtained by multiplying density and specific heat to the measured thermal diffusivity.
The Li2TiO3 pebbles with high sphericity were prepared by slurry droplet wetting method. The average diameter of pebbles was about 1.1 mm. The sample container was specially designed for the pebble bed measurement, which was made of graphite to withstand high temperature. The quartz disc and ring were adopted as a thermal insulator to avoid heat transfer from pebble bed to graphite container. The volume of pebble bed was about 1.7 mL, which was relevant only 3.0 g of Li2TiO3 pebbles. The packing factor of Li2TiO3 pebble bed was about 59.3 %. The thermal diffusivity of the pebble bed was measured in flowing helium at elevated temperature to 800 oC with 100 oC of intervals. The effective thermal conductivity of Li2TiO3 pebble bed was calculated at about 1.4 W/mK at 500 oC ~ 700 oC. The detailed results and trends of thermal conductivity on Li2TiO3 pebble bed will be introduced in the present study.

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P2.053 Engineering design of Wendelstein 7-X alkali metal beam diagnostic observation system

On Wendelstein 7-X a Sodium beam emission spectroscopy (BES) diagnostic system has been installed in 2017 in order to measure plasma edge density and turbulence. The diagnostic setup consists of two parts: an alkali beam injector and an observation system through which we can observe the light emission by the alkali beam.
The observation system consists of two parts, which operate in parallel: a high sensitivity Avalanche Photodiode (APD) camera and an overview CMOS camera. The collected light is divided by a 45° angled mirror, which transmits 2.5% of the light to the CMOS camera while 97.5% goes to the APD camera. The light is focused by an optical lens system and transmitted to APD camera by optical fibres arranged along the observed beam.
To achieve sufficiently high photon flux the APD branch contains relatively big lenses (up to 244 mm diameter) therefore a robust lens holder structure had to be fixed onto a 184 mm diameter window flange at the top of the cryostat. The 45° angled mirror had to be exactly positioned to the centre of the optical axis while its holding structure should not cover the way of the light towards the APD branch. Beam light emission occurs at two identically intensive spectral lines separated by 0.5 nm around 589 nm. A Carbon II line is located exactly on one of the Sodium lines, therefore cutting the Carbon emission was possible by measuring only one of the emission lines. A custom designed filter is used, which can be temperature tuned so as at room temperature it transmits both Sodium lines, while at about 60℃ the Carbon line and one of the Sodium lines is strongly cut.
Engineering design solutions and technical developments of the W7-X alkali BES observation system are discussed in this paper.

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P2.054 Calibration and test of the 6LiF-diamond detector for the HCPB mock-up experiment at JET

Within EUROfusion WPJET3 programme, the unique 14 MeV neutron yields produced in the scheduled JET DT campaign will be exploited to validate codes, models, procedures and data currently used in ITER design in order to reduce the related uncertainties and the associated risks in the machine operation.
One relevant experiment selected for DT is the irradiation of the HCPB-TBM mock-up of ITER which will be located in front of the JET horizontal port of Octant-8. The objective is to take advantage of the significant 14 MeV neutron emission to produce a measurable quantity of tritium to validate tritium breeding predictions in a reactor-like environment. The neutronics performances of the HCPB-TBM will be measured through independent measurement techniques and the experimental quantities compared to the results of MCNP simulation. Of the main importance is thus the measurement of the tritium production rate in the HCPB-TBM. This will be attained by using a single crystal diamond detector (SCD) covered with a thin layer of 6LiF (LiDia detector). The tritium production is obtained by measuring the 6Li(n,α)T reaction rate. This technique was already proved to be effective, however it requires the accurate knowledge of the mass of 6Li.
In the present work the 6Li mass calibration of the selected LiDia detector was performed in a well characterized thermal neutron flux spectrum. Furthermore, a performance test of the LiDia detector was carried out by locating it inside a small polyethylene phantom which was irradiated under 14 MeV neutrons at the Frascati Neutron generator (FNG). The tritium produced by the 6Li(n,α)T reaction was measured. The experimental set-up was accurately simulated by using the MCNP6 code and the calculated tritium production compared to the experimental one. The paper reports the thermal calibration procedure as well as the results of the test experiment.

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P2.055 Nd:YAG Lasers for ITER Divertor Thomson scattering

LD pumped Nd:YAG lasers developed for ITER Divertor Thomson Scattering (DTS) diagnostic can operate with high power 3ns/(1–2)J/(50–100)Hz at wavelengths 1064nm and 946nm.
1064nm Nd:YAG laser technology is well established and, when creating the laser, we are mainly focused on quality of the laser radiation. The laser beam quality is usually affected by thermally induced lens and birefringence in laser rods. Owing to the use of SBS Phase Conjugation mirror to compensate for the aberrations flat-top output beam with a nearly perfect beam quality was achieved in a wide range of output power.
946nm laser with similar parameters is a challenge solution, because the achieved laser power exceeds characteristics of existing lasers operating at this wavelength by more than 10 times. It is lasing on quasi-three-level transitions, where a population inversion can be achieved at high pump intensity only. Conventional MOPA optical schemes with Q-switched master oscillator are not applicable here, because the stimulated emission cross section for 946nm is 5 times smaller than for 1064nm. The ring cavity regenerative power amplifier with relay optics provided a zero diffraction length and image-rotation, which allowed obtaining output energy of 1×50Hz at 3ns pulses with beam divergence of 1.25×DL.
Stable properties of the beam quality of the developed lasers are extremely important for ITER application, since it determines predictable absence of breakdown of all optical elements of complex relay system used to deliver laser radiation on a distance of 40 meters into the divertor plasma. The laser beam quality meant low beam divergence, absence of hot spots on optical elements, reproducible and close to a Gaussian temporal shape.
This paper deals with approaches used under the laser design development and with the detailed properties of the laser operation. The state of the art and outlook of further developments are also presented.

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P2.056 On the use of rhodium mirrors for optical diagnostics in ITER

The first mirrors of optical diagnostics in ITER are exposed to high radiation and fluxes of particles which escape the plasma, in the order of 10^20 m-2 s-1. They are thus the most vulnerable optical component in the optics chain inside port plugs, being subject to erosion, especially by fast charge-exchange neutrals, or to deposition of impurities at flux rates which can reach 0.1nm/s. The material selected for the reflecting surface must combine, among others, a high optical reflectivity in a wide spectral range and a sufficient resistance to physical sputtering – both during normal operation and during cleaning discharges, if any is installed. Rhodium (103Rh, atomic no. 45) was identified early as a possible or even promising candidate.

Rhodium combines several attractive properties, for instance a mass which leads in most cases to low sputtering yields (cascades of binary collisions) together with an optical reflectance (~75%) which is much higher than for some heavier elements considered (molybdenum or tungsten) and, unexpectedly, fairly insensitive to large temperature changes. It is rather inert and its low oxidation is an appreciable advantage in case of steam ingress events (e.g. water ingress from leaks).

Whereas the original diagnostic baseline for the core CXRS diagnostic in ITER (upper port plug No.3) was relying on Mo, we have now turned to Rh as a preferable option which allows for an integrated design in the sense that a cooled holder and an embedded cleaning system, should it be decided for, seem to be feasible. The authors aim to procure monocrystalline rhodium which would prevent to a large extent the diffuse reflection to increase with the damage due to physical sputtering.

Pros and cons of the use of rhodium for a first mirror will be presented with respect to optical, mechanical and robustness aspects and achievements.

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P2.060 Strategy and guidelines for the calibration of the ITER radial neutron camera

A calibration procedure is proposed for the entire lifetime of the ITER Radial Neutron Camera (RNC) diagnostic. The proposed calibration is divided in different phases: pre-delivery calibration, post delivery calibration and periodic calibrations at different time intervals. The RNC calibration relies exclusively on radiation sources available at the domestic agency in the pre-delivery phase and on embedded neutron sources for all the calibrations after the delivery to and installation in ITER. No in-vessel calibration using external neutron sources is required: instead, it is proposed to rely on reference ITER pulses for the tracking of the calibration in time. The use of a neutron generator inside the RNC for neutron efficiency monitoring has been investigated based on present-day commercially available manufacturer, and disregarded as it introduces unnecessary complexity to the RNC diagnostic without substantially improving the long term calibration. The calibration of the RNC will be supported by numerical simulations for the validation of the calibration and for their subsequent monitoring. Radiation sources for each type of detector have been identified
but their strength has not been determined as this will depend on the level of background radiation at the detector location which can be estimated only in the final design phase of the RNC. Monte Carlo calculations will be required for this step and for determining the effect of the embedded calibration sources intended for one type of detector on the other detector types.

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P2.064 First heat flux decay length estimation in WEST with embedded thermal measurements

The main objective of WEST is to study the behavior of the ITER like Plasma Facing Components (PFCs) and to test the resistance and ageing of these components under high heat loads. To achieve these objectives, two independent thermal diagnostics have been developed and installed in the lower divertor of the WEST tokamak. The first one is based on 20 thermocouples embedded at 7.5 mm from the surface in non-actively cooled W-coated graphite PFCs and distributed at different poloidal and toroidal locations. The second one is the Fiber Bragg Grating (FBG) diagnostic. We have installed four optical fiber temperature probes in the lower divertor of WEST, each of them including 11 FBGs equally spaced by 12.5 mm in the poloidal direction (which is equivalent to one ITER-like tungsten monoblock). The probes are embedded in non-actively cooled W-coated graphite PFCs at 3.5 mm and 7 mm below the surface (2 fibers at each depth).

A short description of the thermocouple and FBG diagnostics and their performances will be presented as well as the first bulk temperature measurements obtained in ohmic and L-mode pulses in the WEST tokamak. A 2D nonlinear unsteady calculation combined with the Conjugate Gradient Method (CGM) and the adjoint state is used in order to estimate the space and time evolution of the surface heat flux based on temperature measurements. The heat flux decay length on the target will be extracted from these estimations. The position of the peak heat flux will be investigated for the two main lower single null magnetic equilibriums foreseen to test the ITER-like tungsten PFCs, with X-point location “close” and “far” from the divertor target.

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P2.065 Analyses and structural integrity estimation of the Divertor Thomson Scattering system

The presentation is focused on approaches and results of simulations and used for loading analyses made for new design of the Divertor Thomson Scattering (DTS) in-vessel equipment, including spatial stress strain state, seismic analysis, electromagnetic analysis as well as the most important load combinations.
The ITER Divertor Thomson Scattering system is designed to provide an instrument capable of measuring the profiles of electron temperature and relative profiles of electron density in the outer divertor plasma.
Finite element model of the construction includes updated DTS components such as outer and inner frames of the diagnostic racks, neutron shield attached to the front rack outer frame and main components of the vacuum vessel lower divertor port which contact with DTS system. Thermal analyses taking into account thermal radiation between surfaces of the front rack construction and walls of the lower port were conducted for the transient normal operation mode with 500 MW of fusion power. Based on thermal analysis results, DTS in-vessel elements stress strain state was analyzed and the most stressed points were listed. Electromagnetic analysis was conducted to determine force and moment values applied to the constructions for the most severe plasma disruption events. Moreover, interface accelerations received from the vacuum vessel lower port during EM scenarios were obtained. Stress strain state and temperature maps due to plasma disruption events were also obtained. Finally, some load combinations of event categories I-IV were explored and the obtained results were analyzed.

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P2.069 Estimation of neutral fluxes on the first mirror of H-alpha diagnostics in ITER

The first mirrors of all optical diagnostics will be exposed to the fluxes of neutrals, mainly D, T and Be, from the plasma in ITER. This can lead to formation of erosion and deposition zones on the mirror surface. The location of the zones will depend on D,T/Be flux ratio and geometry of the cutouts in the diagnostic shield module (DSM). In H-alpha diagnostics, the first mirrors in equatorial port plugs are located behind 7 mm diameter entrance pupil (pinhole) to reduce the neutral atom fluxes to the mirror surface.
In this work, the fluxes of neutral D, T and Be on the first mirror of H-alpha diagnostics in the equatorial ports were calculated with Zemax OpticStudio. Only stationary modes of ITER were considered. The geometry of the DSM cutouts and the first mirror units (FMU) were taken from the ENOVIA database. It was shown that the most of the first mirror surface is occupied by an erosion zone with erosion rate about 50 nm/year. The deposition zone is located on the edges of the mirror with deposition rate ~100 nm/year. Being made of single crystal molybdenum, the mirrors will keep their optical properties under erosion. The contaminated edges are not important for the light collection and imaging of the SOL emission in the areas of interest at the ITER first wall within the selected field-of-view.
Resuming, the first mirrors of H-alpha diagnostics will be capable to provide acceptable optical performance in the stationary modes in ITER lifetime. The DC plasma discharge mirror cleaning system is implemented in the FMU design in order to restore the performance if the accidental events cause the FM contamination e.g. due to ELMs or water leak into the vessel.

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P2.070 Irradiation test of fiber optics for H-alpha diagnostics in ITER

The reliability of the optical diagnostics in ITER critically depends on radiation resistance of the fiber optics for a transmission of plasma light to remote detectors. The design of H-alpha diagnostics includes fiber bundles about 60 m long between the port cell and the diagnostic room. The first 10 m of the bundles run through the gamma-neutron fields. This part of the bundle will accumulate the absorbed dose of no more than 2 kGy (for silica) and the total neutron fluence of 1014 n/cm2 in the ITER lifetime (2·107 s or 40 000 pulses). The modern types of radiation resistant silica-based fibers can be applicable in ITER for use in the visible range. The convenient way to select the fiber type is to make in-situ gamma irradiation test. The test has to be performed at a dose rate comparable to the radiation conditions in the ITER facility. The report is devoted to the irradiation tests of silica fiber samples in a Co-60 gamma source at the dose rate of 0.01 G/s. Three types of silica based fibers were tested. The transmission losses of the fibers were measured in-situ in the spectral range of 400-700 nm. The conclusion was made about the applicability of the one tested fiber type for the H-alpha diagnostics in ITER.

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P2.213 Analysis of radionuclidic purity of medical isotope production with d-Li neutron in A-FNS

We are carrying on design activities of an advanced fusion neutron source (A-FNS) in Japan. A large amount of neutrons are produced by Li(d,n) reaction bombarding a 40 MeV deuteron beam of 125 mA with a liquid Li target at the A-FNS. In the Li(d,n) reaction, there are reaction processes with strong angular dependence such as proton stripping and ones with weak dependence such as evaporation. Neutron spectrum depends on locations in the test cell of the A-FNS, especially in high energy region. We are studying the production of medical radioisotope Mo-99 which is the parent nuclide of Tc-99m. There are two reactions in production of Mo-99: Mo-100(n,2n)Mo-99 and Mo-98(n,g)Mo-99 reactions. Mo-100(n,2n)Mo-99 reaction is induced by neutrons above about

8.4 MeV. Mo-98(n,g)Mo-99 reaction is mainly induced by lower energy neutrons. We install the Mo-100 and Mo-98 samples at different locations in the test cell for the suitable production of Mo-99. We calculated neutron spectra in the test cell by using a Monte Carlo transport code MCNP5 with an extension McDeLicious-11 and nuclear data library FENDL-3.1d. Amounts of radioisotopes produced in Mo samples were calculated by using the spectra and an activation code FISPACT-2010. In these radioisotopes, only Tc-99m can be applied for medical use. Radionuclidic purity of Tc-99m is severely required. Tc isotopes other than Tc-99m out of these radioisotopes become a problem in enhance of the purity after chemical separation. From the calculation results, it was found that we could control the amounts of the Tc isotopes by using the isotopically enriched Mo-98 and Mo-100 or natural Mo samples with the right selection of location in test cell which is to say neutron spectrum and cooling time to become the purity higher. From this study, we clarified that the purity of Tc-99m met the medical demand.

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P2.078 AC loss assessment for fusion conductors

The coupling currents loss for fusion conductors is frequently assessed applying a sinusoidal field sweep of fixed, small amplitude and variable frequency. From the initial slope of the loss curve, the coupling loss time constant is derived and applied in the loss calculation over the whole range of field transient. In this work, the traditional AC loss assessment is compared to an alternative experimental assessment made on trapezoidal field change, which mimics the actual field change rate in the central solenoid of tokamaks during the plasma start-up. Dedicated test campaigns in SULTAN provide the data for the assessment. The impact of the new AC loss assessment allows a more realistic prediction of the thermal-hydraulic behavior of the Central Solenoid in operation. The details of the assessment procedure are discussed and a new standard AC loss test procedure is proposed.

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P2.210 Neutron induced Primary knock on spectra and displacement damage on fusion reactor materials (W Fe Cr Cu& Al) at energies up to 14.1 MeV energy

Interactions of 14.1 MeV energy neutrons with fusion reactor materials will result into the production of energetic recoils (primary knock on atoms) that lead to displacement damage in the reactor materials. These neutrons will yield different recoils atoms species of different energy and mass based on different reaction channels. Prediction of displacement per atom (dpa) requires energy spectra of PKA from all probable reaction channels. In the present work, PKA spectra have been calculated for neutrons induced reactions on stable isotopes of Tungsten, Iron, Chromium, Copper, and Aluminum at up to 14.1 MeV energy from all possible reaction channels with optimized nuclear model parameters. PKA spectra are later used in NRT (Norgett Robinson and Torrens), BCA (Binary collision approximation) and BCA+MD (molecular dynamics) approach to calculated the displacement cross section. Reaction cross section of different reaction channels and energy spectra of emitting neutrons, protons and alpha particles are calculated and compared with the existing experimental data from EXFOR data library to select the best fitted nuclear models and parameters. All the reactions modes namely compound nuclear, pre-equilibrium, direct and multiple emission mechanisms have been taken into consideration for cross-section calculations. TALYS-1.8 code has been used for the cross-section calculations and IOTA code has been used for the BCA and BCA+MD simulations.

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P2.081 Reactive power compensation for the pulsed power supply of ASDEX Upgrade

One of the biggest challenges for a fusion reactor with magnetic confinement is the controlled removal of the heating power. ASDEX Upgrade (AUG) is one of the leading experiments in this area and investigates integrated solutions that combine high heating power and wall materials suitable for reactors. To increase the normalized output power towards the values intended for ITER and DEMO, the present AUG program combines the rise of the central heating power by ECRH, ICRF and NBI with an enhancement and optimization of the power supply installation.
To ignite the plasma and to ramp up the currents in the poloidal field coils, high DC voltage is required. During plasma flattop the ohmic losses of the copper made coils have to be covered, only. This part-load operation results in high reactive power demand of the high current converters. The power supply of AUG relies on flywheel generators with limited power and energy. To optimize the energy consumption and to reduce the current load to the generator, a 90 Mvar static reactive power compensation (RPC) has been installed at flywheel generator EZ4. In a first step, 30 Mvar of reactive power have been connected to the generator in order to check the system stability of a converter driven, self-excited generator. After successful testing, the RPC has been extended and operates now with 2x 15 Mvar and 2x 30 Mvar switched compensation modules.
The paper describes the motivation and effort to increase the pulse power capabilities of AUG. It provides details on the design, installation, commissioning and the present state and experimental findings of the RPC system.

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P2.083 The starting mechanism analysis of ITER PF AC/DC in series converters

The International Thermonuclear Experimental Reactor (ITER) Poloidal Field (PF) AC/DC in series converters are composed by three converter units in series to supply megawatt energy to PF coil. With the characteristic that high power, complex operating modes, large amount of snubber capacitors and stray inductances, any inappropriate starting mechanism could introduce over-voltage and then damage the converter. Therefore, it is imperative to give full consideration to the starting mechanism of the ITER PF AC/DC in series converters. The external bypass that located at the downstream of the dc reactors is designed to provide the protection of converter in fault conditions at very beginning, which also provide another alternatives to the starting mechanism of ITER PF AC/DC in series converters. In this paper, the starting mechanisms which assisted by the external bypass is proposed to activate the ITER PF AC/DC in series converters securely and reliably. The starting over-voltage of three starting mechanisms during full-voltage starting are simulated to propose the optimal starting mechanism. According to the simulation, the starting mechanism that start the two converter units firstly and then start the third one is selected as the optimal starting mechanism for ITER PF AC/DC in series converters. The feasibility of the proposed optimal starting mechanism is verified by the experiment on the ITER PF converter sequential control test facility.

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P2.104 PFC Heat Flux Enhancement from Magnetic Field Errors in the NSTX-U Recovery Project

The Plasma Facing Components (PFCs) of the National Spherical Torus Experiment Upgrade (NSTX-

U) protect the vacuum chamber wall from high plasma heat fluxes which are mostly carried by energetic particles that flow along magnetic field lines. The magnetic field lines will have very shallow impingement angles to the PFC surfaces (as small as 1o), a consequence of flux expansion at the divertor. The heat flux along a shallow field line can be more than 50 times greater than the nominal axisymmetric heat flux making the surface normal heat flux very sensitive to field errors arising from coil misalignments and distortions.

Low aspect ratio spherical tokamaks place the divertors at a radius closer to the centerline of the machine and thus are more affected by misalignments with the toroidal field. The inner coils of NSTX-U are very close to the plasma facing surfaces of the divertor and thus small coil position errors have a large effect on the incident angles of the plasma on the tiles.

Adding to this heat flux enhancement is the need to protect leading edges of individual tiles exposed by the gaps between tiles (needed for installation and thermal expansion) and protrusions toward the plasma introduced by tolerances. NSTX-U has chosen to taper (aka fishscale) the tiles in one direction toroidally to provide shadowing of adjacent tiles. This protects tile edges at the cost of further increasing the surface normal heat flux.

An analysis of the influence of each potential perturbation (i.e. displacements, rotations and distortions) of each PF and TF coil along with tile tolerances on surface heat flux enhancement was undertaken to provide a tool for evaluating their relative importance. The combined impact was assessed using Monte-Carlo analysis. This in turn was used to provide guidance for coils and PFCs fabrication and assembly tolerances.

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P2.085 Type Tests of Counter Pulse Circuits for the ITER Fast Discharge Units

The ITER magnet system comprises 30 main superconductive coils, which will create steady-state and slowly varying magnetic fields with a total energy up to 50 GJ. The fast protective discharge of this energy in case of a quench (superconductor-to-normal transition) is provided by closing the coil current circuits with the help of protective make switches (PMS) and inserting discharge resistors (DR) into these circuits by opening circuit breakers forming part of fast discharge units (FDU).
Two FDU systems have been designed, namely, TF FDU for protection of the toroidal field (TF) coils with constant current direction and PF/CS FDU for the poloidal field (PF) coils and the central solenoid (CS) modules, where the current direction and magnitude might vary during operation cycles.
One of the main parts of each FDU is a counter-pulse circuit (CPC) consisting of one or two counter-pulse units and a charger. The CPC units are intended to provide a current pulse in vacuum interrupters forming part of the two-step FDU circuit breaker in the direction opposite to the coil current direction. The TF FDU includes two unipolar CPC units (CPC-U) with output circuits connected in parallel with each other. The PF/CS FDU has one bipolar CPC unit (CPC-B), where the direction of output current pulse can change depending on the direction of the coil current.
The paper describes the designs of two types of the CPC units and the results of the type tests carried out on their full-scale prototypes. The successful results of the tests demonstrated the working efficiency of the CPCs, confirmed their compliance with the design requirements and served as the basis for beginning of manufacturing of these devices to be delivered to the ITER Site.

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P2.089 Phase-detection-based feedback control for the power supply in tearing mode control system on J-TEXT

Tearing Mode (TM) creates magnetic islands in the tokamak, which will cause mode locking and major disruption. A new method for applying modulated magnetic perturbation is explored to suppress magnetic island and accelerate island rotation by using external resonant perturbation (RMP) coils in the J-TEXT tokamak. The phase difference between TM and external RMP is denoted by Φ. RMP has a stabilizing (destabilizing) effect on island when $0.5\pi<\phi<1.5\pi$ ($-0.5\pi<\phi<0.5\pi$) and an accelerating (decelerating) effect when $\pi<\phi<2\pi$ ($0<\phi<\pi$). Moreover, a net suppression effect has been proved by numerical simulation result when $\pi<\phi<2\pi$.
To verify the mechanism above, a variable frequency current-pulse power supply has been developed to suppress TM, which has the maximum frequency of 5 kHz and current of 3 kA. The phase-detection-based feedback control system is designed to control the power supply properly. The feedback control system contains two parts, TM phase detection and power supply control. The TM phase information is identified from the poloidal array of Mirnov probes. A phase prediction algorithm is developed to eliminate the phase difference between TM phase and actual RMP phase, which is caused by the delay of current rising time. The digital filter and the zero-crossing error correction algorithm are developed to make TM phase detection more accurate. The power supply control can achieve two kinds of control mode, bus voltage close-loop control and output current close-loop control. And the dynamic output current amplitude control is developed to adjust the flat-top of the current-pulse when the frequency of output current changes in a wide range in a short time. The system has been used in the experiment in the J-TEXT tokamak and has a good result.

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P2.095 Design of the control system of acceleration grid power supply for CFETR N-NBI prototype

In order to realize the 200keV negative-ion-based neutral beam injector (N-NBI) prototype of China Fusion Engineering Test Reactor (CFETR), an acceleration grid power supply (AGPS) is under development. The AGPS adopts single stage inverter type topology and is rated at 200kV/25A/3600s. Here, the paper presents a control system that can fulfill the high requirements in both steady state and beam modulation state of AGPS performance.
Based on the PCI extensions for instrumentation (PXI) technology, the control system can be well integrated and communicate with upper control system of the N-NBI prototype. The control system is designed to function correctly and reliably in real-time control, supervisory control and protection of AGPS. Furthermore, the control system can also identify the frequent breakdown between acceleration grids and unpredictable beam off correctly within 50 us, which correspond to short-circuit of the load and a loss of load respectively. At the same time, simulation verifies that the control strategy can limit the voltage ripple and overshoot, respectively, within the range of ±5% and 15% of nominal outputs.
In addition, a 10kV/6A AGPS prototype has been realized, showing the availability of the control system in terms of operating sequence, control logic, and output adjustment, especially its response to a fault or abnormal conditions of AGPS as well as the breakdown and beam off. Restart operation after breakdown occurring is also tested on the prototype. Thus, the presented control system satisfies the requirements of the 200kV/25A AGPS.

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P2.218 Progress in the design development of EU DEMO Helium-Cooled Pebble Bed primary heat transfer system

In the frame of the activities promoted and encouraged by the EURO-fusion Power Plant Physics and Technology (PPPT) department aimed at developing the EU-DEMO fusion reactor, strong emphasis

has been recently posed to the whole Balance of Plant (BoP) which represents the set of systems devoted to convert the plasma generated thermal power into electricity and to deliver it to the grid. Among these systems, a very important role is played by the Breeding Blanket (BB) Primary Heat Transfer System (PHTS) as it is responsible to extract more than 80% of the fusion plasma power.
In this framework, University of Palermo, Ansaldo Nucleare and CREATE have focused their work to improve thermal-hydraulic, safety and integration features of the Ex-Vessel PHTS for the Helium-Cooled Pebble Bed (HCPB) BB concept of DEMO.
Starting from the outcomes obtained last year that have allowed to highlight some criticalities which needed design changes, the paper describes progress and developments of the HCPB PHTS as well as the goals which have been achieved. The results of the research activity carried out show, in fact, an overall increase of both safety characteristics and thermal-hydraulic performances since a reduction in total coolant inventory and total pressure drop has been respectively reached. Nevertheless a critical assessment of these key parameters reveals that some issues are still open in terms of design integration and feasibility of the whole DEMO BoP for the helium-cooled blanket option, indicating that additional efforts are required to make this technology more attractive.

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P2.100 Demonstration tests of mechanical lap and edge joints of 10-kA-class STARS conductors

Segment-fabrication of high-temperature superconducting (HTS) magnet is an attractive concept to solve engineering issues of helical fusion reactor having huge and complex superconducting helical coils. There are two designs as the segment-fabrication: 1) Remountable (demountable) coil option; the entire half-pitch helical coil segments are connected with demountable multi-conductor joints, 2) "Joint-winding" option; one helical-pitch single conductor segments are connected with once-through joints. In those designs, HTS conductors consisting of simply stacked REBCO (Rare-earth Barium Copper Oxide) tapes embedded in copper and stainless-steel jackets are used and the conductor is named STARS (Stacked Tapes Assembled in Rigid Structure) conductor. We have developed mechanical edge joint and bridge-type mechanical lap joint of STARS conductors for the former and the latter designs, respectively. In our previous study, we have already achieved 80 nano-ohms using the mechanical edge joint of 1-kA-class STARS conductors at 77 K. The next target of the edge joint is achieving 1 nano-ohm using 10-kA-class conductors at 4.2 K. Another study has also achieved 2 nano-ohms using the bridge-type mechanical lap joint of 100-kA-class STARS conductors at 4.2 K. However, it took over a half day to fabricate the joint because the joint piece was not integrated but just individual REBCO tapes.
In this study, we fabricated mechanical edge joint and bridge-type mechanical lap joint of 10-kA-class STARS conductors to demonstrate their low joint resistance and quick fabrication compared to previous studies. An integrated joint piece was newly introduced to shorten the fabrication process of bridge-type mechanical lap joint, which might make the fabrication time to be less than 3 hours per one joint. The joint resistance was evaluated at 77 K and 4.2 K using liquid nitrogen and liquid helium cryostat. The detail of the conductor geometry and results are presented.

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P2.102 Layered W-WC composites prepared by FAST

One of the main difficulties of designing fusion reactor is the development of plasma-facing materials that have to be resilient to the proximity of plasma. Pure tungsten is a primary candidate for this material but has to be strengthened either with particles or fibers to improve its' brittleness at moderate temperatures and inhibit recrystallization as well as grain growth at higher ones.
To limit the W grain growth, we proposed the incorporation of tungsten carbide particles in tung-

sten matrix. Samples were prepared from W and WC powder mixtures and consolidated with field assisted sintering technique (FAST). Layered W-WC composites with the gradually increased content of carbide phase were prepared to tailor the thermal conductivity of the material for monoblock of divertor. The layering will reduce thermal shocks and control heat transfer to the copper-based cooling system.
We have already confirmed, that sintering of W and WC particles by FAST induces in-situ high-temperature reaction with the final composition of cubic W with hexagonal W2C phase. W-W2C composites exhibit high density and improved mechanical properties at room and elevated temperatures. If the W2C content in the composite is 5 wt % or higher, W-grain growth is inhibited even after aging at 1600℃ for 24 h, due to the pinning of W grain boundaries with smaller W2C. The mechanical properties of aged samples did not impair, and chemical composition remained unchanged. Layered composites with the gradually increased content of W2C was successfully prepared. If the difference in the carbon content between two layers is too high, the growth of elongated W2C grains is induced. XRD and EBSD analyses confirmed that these elongated W2C grains had preferred orientation (0,0,2). The microstructures of such phases were thoroughly examined.

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P2.105 Erosion and deposition investigations on Wendelstein 7-X first wall components for the first operation phase in divertor configuration

In the Stellarator Wendelstein 7-X with its twisted 3D magnetic field geometry, studies of material migration with respect to first wall components becomes very important in view of the envisioned long-pulse operation. A variety of erosion/deposition probes were installed on graphite plasma-facing components exposed at three different nominal heat load levels between 0.1 and 10 MW/m2. At the location of the highest heat loads, 18 exchangeable divertor target elements were coated with C/Mo marker layers on graphite and for the lower loads, 44 Si-wafer probes were installed. An extensive array of >30000 probes in form of amorphous carbon coated Ti-Zr-Mo screw heads was installed around the torus capable to withstand ≤ 0.5 MW/m2.

After the first successful operation phase in divertor configuration (OP1.2a), all the probes at higher and lower load levels were removed, out of 30000 screws about 350 have been exchanged at selective locations along the toroidal and poloidal directions. The removed probes are being currently analysed by different methods: Rutherford backscattering, scanning electron microscopy, laser-induced breakdown spectroscopy, nuclear reaction analysis etc. The deposition profile of carbon layers along the whole vessel wall is being measured using a colour analyser. A complex pattern of erosion and deposition zones can be identified optically with coloured deposition layers after about 60 plasma minutes excluding the glow discharge cleaning. First tests with tungsten were performed on short samples using mid-plane manipulator. Further analysis of the composition will be presented.

For the operation phase 1.2b a new set of probes are being installed, 16 Si-wafer probes are being replaced with 12 material probes providing directionality information for the deposited materials and 4 cavity probes. Further tests with tungsten will be performed by replacing 21 heat shield graphite tiles with tungsten coated ones and additionally coating 3 tiles on an exchangeable target element.

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P2.108 Fracture mechanics analyses of divertor vertical target under thermal loading conditions

In-vessel plasma facing components such as first-wall, blanket and divertor modules should with-

stand harsh design conditions. In particular, since the divertor module undergoes extreme thermal loads, several tests for mono- and multi-block mock-ups as well as lots of stress analyses for the mock-ups and module themselves have been carried out. However, there were a little fracture mechanics studies limited on cracked simple geometries subjected to sudden severe thermal loads during micro-seconds period due to edge-localized mode(ELM). In this study, parametric linear elastic fracture mechanics assessment was conducted to determine critical crack lengths(CCLs) of ITER divertor vertical target. Typical heat fluxes under normal steady-state and ELM conditions were applied to the target with a postulated semi-elliptical surface crack on the tungsten blocks. Systematic finite element analyses were performed by changing crack locations, orientations and depths with a constant aspect ratio of 3. Stress intensity factors(SIFs) were computed and compared with the corresponding plane strain fracture toughness in ITER SDC-IC. Effects of crack features and heat fluxes on CCLs of the divertor vertical target were fully discussed.

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P2.110 Thermal properties of stabilized jets for the liquid metal divertor REVOLVER-D

The thermal property of the jet stabilized by an internal flow resistance has been investigated in order to apply the liquid metal divertor consisting of molten tin shower jets named the REVOLVER-D, to the helical fusion reactor, FFHR. The allowable heat load on the REVOLVER-D is higher than that of conventional solid divertors. The droplet formation of the jet can be avoided by inserting an internal flow resistance like chains or wires. On the other hand, the vapor from liquid metal can deteriorate the core plasma performance. The temperature of tin must be kept less than 1000 K to suppress the vapor pressure less than 10-5 Pa although the vapor pressure of molten tin is lower than that of other metals. The temperature increase of jets can be suppressed if the fluid is uniformly mixed. However, jets have temperature distribution and the peak temperature can exceed 1000 K. The thermal property needs to be investigated to control the temperature of molten tin jets. Therefore, experiments with the U-alloy78 circulation device and numerical simulations with ANSYS have been carried out. U-alloy78, an alloy of tin, bismuth, and indium is selected as the working fluid instead of molten tin because of low melting temperature. The density, dynamic viscosity, and specific heat of U-alloy78 are similar to those of molten tin. The effect of heat transfer enhancement by the turbulence generated by the internal flow resistance is also investigated. The dependence of the thermal property of the jet on the shape and size of the internal flow resistance will be given in this presentation.

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P2.223 Validation and sensitivity of CFETR design in EU systems codes

The Chinese Fusion Engineering Test Reactor (CFETR) bridges the gap between ITER and a demonstration fusion power plant (DEMO). The primary objectives of CFETR are: demonstrate tritium self-sufficiency, ~ 1 GW fusion power, operate in steady-state and have a duty cycle of 30-50 %. CFETR is in the pre-conceptual design phase and is currently envisaged to be a two-phase machine (phase I ~ 200 MW, phase II ~ 1 GW).

In 2016 the EU and China began a collaboration on topics relating to nuclear fusion research. This contribution documents the progress on the collaboration on systems codes. Systems codes attempt to model all aspects of a fusion power plant using simplified models (0-D, 1-D) and capture the interactions between plant systems. This allows the user to explore many reactor designs at a high level and optimise for different figures-of-merit (e.g. minimise major radius). The EU systems codes used for the work are PROCESS and SYCOMORE.

This paper details the work on analysing the 2018 CFETR design in EU systems codes and the feasibility of the design with regards to meeting the performance objectives and operation of the machine. The work considers the two-phased nature of the device and comments on the systems codes output for phase I and phase II. In combination with systems codes, uncertainty
quantification tools are used to quantify the sensitivity of the CFETR design to the input assumptions in the systems codes. This paper details the key sensitivities for the CFETR design and whether at the bounds of the uncertainty CFETR still fulfils its high-level objectives.

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P2.111 Thermal electromagnetic and structural analysis of gas baffles for the TCV divertor upgrade

As part of an ongoing divertor upgrade of the TCV tokamak [1] it is planned to add gas baffles on the inner and outer wall of the vacuum vessel to form a divertor chamber of variable closure. The baffles promise to increase the compression of neutral particles in the divertor and, thereby extend the divertor research on TCV towards more reactor relevant, highly dissipative divertor regimes.
It is foreseen to construct the baffles entirely of polycrystalline graphite, also used for the protection tiles in TCV. The thermal analysis of the baffles considers exposure to heat loads expected both during normal operation and off–normal events. During normal operation even an extremely large extension of the power carrying plasma channel towards the baffle for the entire duration of a TCV discharge of 2 seconds does not give cause for concern with temperatures increasing not more than 400 K. An analysis of off–normal events identifies an accidental limiting on the outer baffle as the worst case scenario, where the peak heat flux reaches 30 MW/m2 resulting in sublimation temperatures of 2200 K in ~0.5 seconds. The electromagnetic analysis considers Halo currents, which can occur during disruptions, as the worst case scenario. It is found that a Halo current of 250 kA will result in an average vertical force in the baffles of up to 620 kN/m3. The worst case scenarios for thermal and electromagnetic loads lead to a maximum tensile stress of 19 MPa and a maximum compressive stress of 90 MPa and, thereby remain below their respective material limits of 37.5 MPa and 150 MPa.
The obtained results of the thermal, electromagnetic and structural analysis validate the proposed solution for the baffles and guide further optimisation of their design.
[1] H. Reimerdes, et al., Nucl. Mater. Energy 12 (2017) 1106.

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P.228 Chemistry and Corrosion Research and Development for Water Cooling Circuits of EU DEMO

The European DEMO design will potentially use single phase water cooling in various components that require protection against corrosion damage. Coolant conditions will be similar to fission PWRs but with additional considerations arising from materials choices (Eurofer-97, CuCrZr), 14 MeV neutron irradiation, the presence of tritium, and strong magnetic fields. Presently, many aspects of the water chemistry and corrosion behaviour are not well defined, and several strands of work, reported here, are ongoing to address these challenges under the EUROfusion framework in collaboration with industrial partners.
The foundation of this work is a review of relevant operating experience from fission LWR plant to understand the potential for technology transfer, supported by radiolysis modelling to assess options

for suppression of oxidising species under high energy neutron irradiation. This has specifically considered the interaction of water with Eurofer-97, utilised in the water-cooled lithium-lead blanket concept. High temperature water corrosion testing facilities have also been employed to expand the corrosion database, reported here an approach using micro-scale samples of structural alloys to study their susceptibility to stress corrosion cracking, and a small-scale flow cell approach for in situ corrosion measurement under changing chemistry conditions.
Consideration has also been given to the effect of intense magnetic fields on corrosion through exposure of Eurofer-97 coupons to a magnetic field intensity of 0.88 T and temperatures of 80℃. Further work is reported aimed at identifying the nature and extent of any impact on corrosion behaviour in higher temperature water.
In-vessel cooling of the divertor will use CuCrZr under lower-temperature, high flow conditions, which will lead to different considerations and the potential for flow assisted corrosion. Additionally, high, unidirectional, heat fluxes lead to a radial temperature profile and the possibility of subnucleate boiling. A separate test set-up, currently under construction, to expand this corrosion database is described.

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P2.114 Effect of carbon impurity reduction on Hydrogen isotope retention in QUEST high temperature wall

The elucidation of hydrogen recycling in plasma facing materials is one of key issues to sustain the steady state plasma during fusion operation. QUEST (Q-shu University Experiment with Steady-State Spherical Tokamak) is operated by only hydrogen plasma with all metal plasma facing wall under higher wall temperature of 473 K. However, a mixed material deposition layers contained with carbon, were formed on the top surface of plasma facing walls after the plasma experiments, which would change the hydrogen isotope retention characteristics. Recently, the amount of carbon was remarkably reduced, and surface structure was clearly changed. Therefore, this study focuses on the effect of mixed material deposition layer on hydrogen isotope retention in large plasma device, QUEST under high temperature wall.
The W samples were installed in the plasma facing walls (Top, Equatorial and Bottom positions) of QUEST and exposed more than 4000 shots of hydrogen plasma in each 2015AW (Autumn-Winter), 2016SS (Spring-Summer) and 2016AW campaign. After hydrogen plasma operation in QUEST, the surface chemical states and morphology were studied by XPS and TEM. The hydrogen retention was evaluated by TDS. In addition, 1 keV deuterium ion (D2+) was implanted to evaluate the enhancement of D retention by hydrogen plasma exposure.
It was found that the composition of elements in the mixed material deposition layers was clearly increased as the plasma experiment was accumulated. Major D desorption temperature for 2015AW Top sample was found at 400 K, but that for 2016SS and 2016AW Top sample was located at around 600 K, indicating that D trapping by vacancies would be the major trapping states and most of dislocation loops were recovered by high temperature wall operation. In the presentation, detail experimental results will be shown and the hydrogen isotope retention behavior in QUEST will be discussed.

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P2.115 Investigation of novel technology application in the high heat load components of ICRF antenna

Plasma must be heated by external heating for ignition in the future fusion reactor, ICRF heating is a favorable high-density plasma heating method since the fast wave launched from ICRF antenna can be transmitted to plasma core even in high-density plasma. With the heating power larger, the

exposed antenna surface enduring heat becoming higher, Faraday shied (FS), as one of the key components, faces plasma and undertake high heat load during plasma discharge. High stress and deformation on the FS is produced without removing the heating, which drastically reduces the antenna performance, even damages FS structure to threate RF antenna safety. To explore high-efficiecy heat transfer enhancement technology to remove heat on the FS, one finite element analysis was used to analyze heat and temperature distribution on FS by different heat transfer ways, optimizing the FS structure. Based on the analysis results, several novel heat transfer enhancement way are proposed. Some workpieces with proposed heat transfer enhancement structure were designed and manufactured for test. Afterwards, two mockups with one quarter dimension of EAST ICRF antenna were manufactured without and with optimum heat transfer enhancement technology based on former test results, they were tested under same high heat load , significant results are obtained and presented in the paper, results are shown that heat transfer enhancement technology are drastically improved heat transfer performance of FS, which lay the foundation to high-power, steady-state operation ICRF antenna design in the future.

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P2.232 Testing of some potential techniques for the DEMO radioactive waste management

Selection of technologies that are considered for detritiation and recycling of DEMO waste materials has been made. To study treatment of waste technological process at DEMO facility where it has to be accounted groups of specific materials as composites, metals, oxides and others. From them the DEMO facility will be constructed and which are considered for further modification of this unit. Nevertheless, the new design changes of DEMO had significant impact on waste and recycling. For example, there would be very different requirements for the different breeding blanket options.
Test information was submitted about the potential MSO and IMCC devices for the part of DEMO plant the Waste Management. The Molten Salt Oxidation (MSO) and the Induction Melting in Cold Crucible (IMCC) facilities has to be flexible for specifying kinds and quantities. The IMCC technology was tested for suitable materials that require temperatures up to 1000℃ above melting point for the desired changes. The MSO technology can apply to different characteristic materials, typically from a liquid or suspension to small solids. For both technologies, groups of characteristic materials were selected, some of the representatives were tested and evaluated from the safety and environmental point of view as well.

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P2.118 On-line measurement and removal of hydrogen isotopes in the plasma-facing material of tungsten

The on-line measurement, removal and recovery of hydrogen isotopes in plasma-facing materials are important issues for Tokamak during long-time discharge operations. The laser induced desorption system (LIDS) was designed and built from the laser induced breakdown spectroscopy (LIBS) system with a quadrupole mass spectrometer (QMS). As one part of the comprehensive ECR plasma system, LIDS can be used for in-situ hydrogen isotope retention analysis. Deuterium was introduced into tungsten by means of the gas-phase hot charging method and the linear plasma method, respectively. The results shows that the amount of deuterium introduced into tungsten by plasma is higher that by the thermal diffusion method. The desorption activation energy (1.04 eV) of deuterium introduced into W by plasma is higher than that (0.81 eV) of deuterium by the gas-phase hot charging method. The deuterium trapped in tungsten can exist steadily for a long time under ambient conditions. And deuterium introduced in W by the gas-phase hot charging method is difficult to be removed, since the depth distribution of deuterium is far greater than that of deuterium introduced by plasma.

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P2.119 Hydraulic analysis of EU-DEMO divertor plasma facing components cooling circuit under nominal operating scenarios

Within the framework of the Work Package DIV 1 - "Divertor Cassette Design and Integration" of the EUROfusion action, a research campaign has been jointly carried out by University of Palermo and ENEA to investigate the steady state thermal-hydraulic behaviour of the DEMO divertor cassette cooling system, focussing the attention on its Plasma Facing Components (PFCs). The research campaign has been carried out following a theoretical-computational approach based on the Finite Volume Method and adopting the commercial Computational Fluid-Dynamic code ANSYS-CFX.
A realistic model of the PFCs cooling circuit has been analysed, specifically embedding each Plasma Facing Unit (PFU) cooling channel with the foreseen swirl tape turbulence promoter, hence resulting in a finite volume model much more detailed than those assessed in previous analyses. Its thermal-hydraulic performances have been numerically evaluated under nominal steady state conditions, also comparing the obtained results with the corresponding outcomes of analogous analyses carried out for a simplified PFCs configuration, without swirl tapes. Moreover, the main thermal-hydraulic parameters have been evaluated in order to check whether the considered PFCs cooling circuit might fulfil the total pressure drop requirement ($\Delta p < 1.4$ MPa), providing a uniform cooling of the Vertical Target PFU channels with a viable CHF margin (> 1.4).
The PFCs cooling circuit thermal-hydraulic behaviour has been additionally assessed at alternative operative conditions, issued to check the viability of a coolant velocity reduction, in order to minimize corrosion and vibrations inside the PFU channels.
Models, loads and boundary conditions assumed for the analyses are herewith reported and critically discussed, together with the main obtained results.

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P2.121 Visualisation of subcooled pool boiling in nanofluids

High-performance cooling is of vital importance for the cutting-edge technology of today, from nanoelectronic devices to nuclear reactors. For fusion reactors, subcooled boiling heat transfer is expected to play a critical role for the safe and efficient operation of components exposed to high heat flux. Recent advances in nanotechnology have allowed the development of a new category of coolants, termed nanofluids, which exhibit superior thermophysical characteristics over traditional heat transfer fluids, including water. The present paper reports qualitative results of an experimental investigation of deionised water and Al2O3-H2O nanofluids under subcooled boiling conditions in a pool boiling cell. The purpose was to visually evaluate the impact of nanoparticles on the bubble dynamics and nucleation site activity at the heated surface—bare NiCr wire. It was observed that when nanoparticles are present, the nucleation site density, bubble size at departure and frequency of bubble generation from the surface of the heated wire are remarkably modified. Intense nanoparticle deposition on the heated wire surface was identified as a key mechanism for the observed differences, which altered surface structure and wettability of the wire. The outcome of this work is a step towards the evaluation of the applicability of nanofluids in cooling applications via boiling heat transfer.

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P2.122 China's technological achievements on ITER enhanced heat flux first wall in the pre-PA qualification

The Chinese Domestic Agency (DA) is procuring ITER Enhanced Heat Flux (EHF) First Wall (FW) panels, representing 12% of the total number of ITER FW panels being procured. The EHF FW panel shall withstand a surface heat load up to 4.7 MW/m2 during ITER operation. Prior to the implementation of the ITER Procurement Arrangement (PA), several key technologies in manufacturing the EHF FW panel have been qualified in China. In the pre-PA task, ITER grade CN-G01 high-purity beryllium has been developed; its chemical composition, mechanical properties and grain size satisfy the ITER requirements, and it has been qualified as a choice of material for the ITER FW PAs. The use of strict quality control in the manufacturing of CuCrZr alloy was aimed at increasing the reliability of the hypervapotron (HVT) concept in the EHF FW fingers, while a minor design improvement on HVT cooling channel helped improve the fatigue performance as shown by thermo-mechanical analysis results. Investigation showed that sputtering Ti/Cu coating on the beryllium tiles helps in reducing possible defects and results in better quality of Be-Cu hot isostatic press (HIP) bonding interfaces. As a result of such measures, China DA completed two EHF FW semi-prototypes with one successfully passing the high heat flux test at 4.7 MW/m2 for 7500 cycles and 5.9 MW/m2 for 1500 cycles. Neither damage to the semi-prototype nor off-normal surface temperature increase was observed, which brought the semi-prototype qualification campaign to a successfully conclusion. China DA then signed the PA in 2016 for production of the ITER EHF FW panels. In the PA implementation, China DA plans to investigate the maximum acceptable beryllium tile size for CN EHF FW panels, as well as the acceptable defect size in the Be-Cu HIP bonding interface, which will help control PA cost and schedule.

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P2.235 Overview of the Methods developed for Fission Plants Safety relevant to the Safety of Fusion Facilities

Safety studies for fusion facilities are commonly conducted using codes originally developed for fission reactor accident analysis and adapted to model the fusion-relevant phenomena. Nevertheless there are many "fission developed" methods still not considered in fusion safety assessment which could offer significant advantages in the fusion power commercialization. Along with solving the safety and licensing critical for the fusion power commercialization will be as well the ability to reduce the cost and increase the efficiency of the power production. Among other means these were achieved in the fission power by limiting or even avoiding the conservatism in safety assessment, by improving the methods and use of the state-of-the art tools. There are many reasons for looking into the fission - similar nuclear regulations, the same nuclear safety principles and limits apply, use of mature and proven methods, etc.
The paper will address the following topics
• Experimental programs, Test Matrixes and Data bases
• Computer codes development, verification and validation
• Computer codes assessments
• Conservative or Best Estimate (BE) methodology
• Uncertainty estimation methods
• Phenomena Identification and Ranking Tables (PIRT)
The parallel between the fission and fusion safety approaches and accident analyses methodologies will be drawn. For each of the above topics a brief presentation of the fission historical development followed by an overview of published adaptations of methods and their applications to fusion safety will be reported. The presentation will draw in particular on availability of qualified tools for accidents analyses, use of PIRT, the verification and validation of computer codes by means of separate and integral effect tests and establishing benchmark problems as well on code assessment and development of multi-physics, multi-fluids integrated code systems. These efforts should be aimed at developing a systematic safety demonstration defined by an integrated fusion safety assessment methodology.

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P2.123 Sublimation of advanced tungsten alloys under DEMO relevant accidental conditions

Future fusion power plants require the development of a first wall armor material withstanding extreme particle and heat loads. Considering safety, the formation of long-lived radioactive isotopes when irradiated with neutrons and a tritium inventory has to be prevented. As tungsten (W) meets these safety requirements, has a low erosion rate, high melting point, and high thermal conductivity, it is a promising candidate for the first wall armor.
There is another important safety consideration: in a loss of coolant accident, cooling systems fail and air ingress into the vacuum vessel may occur. Due to the nuclear decay heat, temperatures in the range from 1200 K to 1450 K for several weeks are predicted in such a scenario: the radioactive W oxidizes and volatilizes, posing a severe hazard for the environment. For a successful future fusion power plant like DEMO, a new material is required. The new material should preserve the advantages of W coupled with suppressed sublimation in case of an accident.
Currently an alloy containing W, ~12 weight % chromium, and ~0.6 weight % yttrium consolidated by Field Assisted Sintering Technology is investigated and shows promising results. During exposures to deuterium plasma, erosion yields are similar to pure W. Moreover, the oxidation resistance is significantly improved: complete mechanical destruction is avoided for at least three weeks.
The major goal is to control the sublimation which is responsible for a release of radioactivity. In this work sublimation rates are measured explicitly for the first time in humid air. WO3 sublimates at a rate of 5 g/m²/h at 1273 K in humid air. At the same conditions the alloy suppresses sublimation by a factor of 35 as compared to that of pure W.
The results and consequences are discussed. Further, the mechanisms yielding the improved oxidation resistance are analyzed and discussed.

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P2.128 New route towards micro-structuring tungsten as stress relieving concept

In order to realize a commercial feasible fusion reactor, the life time of plasma facing materials (PFM) and components (PFC) is one of the key issues. Steady state loads in the range of 10 MW/m² and in addition millions of transient events with 0.6 - 3.5 GW/m² represent a huge challenge and lead to severe damages. As first wall, tungsten armor on a low activating structural steel is planned. However, the difference of their thermal expansion coefficients causes large thermal stresses during operation and the cyclic heat loads can lead to a detachment of the tungsten armor. In addition, surface roughening and cracking caused by the repeating incident of transient events can drastically increase the erosion rate. In order to reduce both types of stresses a promising new concept and first results are presented.
Tungsten wires (Ø between 100 µm and 200 µm), which are relatively cheap and easily available from the light bulb industry, are coiled on a square shaped spool. Cutting the edges into slices of different thicknesses, dozens of samples were successfully produced at once. Slices of 1 mm thickness placed in between steel and bulk tungsten can drastically reduce the thermal stress due to the flexibility provided by the ductile wires. At the same time a similar heat conductivity like bulk tungsten is achieved in direction of main heat flux. Supporting FEM simulations are presented. Taking this idea further, the possibility to use 5 mm thick slices directly as PFM is investigated. As the typical crack distance after transient loading is in the range of some hundred µm, wires with smaller diameters could withstand the stresses, reducing the damage to the surface. With high heat flux tests on wires vs. bulk we show the high potential of this concept.

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P2.131 Reviewed design of the high heat flux panels for the AUG and W7-X neutral beam calorimeter

Each of the neutral beam injectors in the experimental devices ASDEX Upgrade (AUG) and Wendelstein-7X (W-7X) can be equipped with up to four positive ion sources with a neutral beam output power of 2.5 MW each. For the conditioning of the system, a movable calorimeter is placed in the path of the neutral beam to dump the heat load. The core of the calorimeter consists of a set of so called calorimeter panels (CP), six for each beam, arranged in defined positions and tilted with respect to the beam axis in order to spread the heat flux density from ~70 MW/m² to a limit below 25 MW/m². The CPs are made out of CuCrZr for its high thermal conductivity together with excellent mechanical strength at higher temperatures. They are intensively water cooled with a mass flow of 3 kg/s each, through a net of many small channels.
The design, originally from 1988, specified the lifetime of the CPs to 25000 heating cycles and discarded deformation due to thermal cyclic loading. This was predicted with the tools available at the time but the experience shows unexpected plastic deformation effects from the beginning. After some years of operation evidence of fatigue is observed. This has led a few times in the past to fatal water leaks and consequently to costly NBI system shutdown for repair. For that reason, careful inspection of the CPs is performed at every maintenance phase and damaged CPs are exchanged for new ones.
The causes leading to the failure and an optimized panel design are presented in this work. The new design aims at minimizing fatigue from thermal cycling by reducing thermal effects, improving material properties and increasing the flexibility of the panel.

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P2.132 A Stereoscopic Eye-in-Hand Vision System for Remote Handling in ITER

The ITER maintenance is done by means of Remote Handling (RH) systems. During maintenance operations, the RH operator is intended to utilize user interfaces for commanding, monitoring and controlling the RH Equipment. The user interfaces are, for example, GUIs, haptic and joystick devices, Virtual Reality (VR) systems and camera views on the RH Equipment and its environment. Many RH tasks involving movers and robotic arms require millimetre accuracy but camera views are often very limited, of poor quality and might be unavailable during the most close and accurate steps. In these cases, the RH operator is constrained to relying on VR providing that the VR models of RH Equipment and its environment be accurately calibrated, but should also take advantage other complementary technologies such as synthetic viewing and computer aided teleoperation. In this context, the purpose of this research was to prototype and evaluate a novel stereoscopic vision software system called 3D Node that locates and tracks the position and orientation of a target item with respect to a stereo-camera attached to the wrist of a robotic manipulator arm. The 3D Node features stereo-camera calibration, target depth mapping, target position and orientation detection and online target tracking including views by overlaying VR to cameras views for quality checks. The tracking information would be valuable for updating VR models and implementing augmented reality and synthetic viewing functionalities. This paper reports the 3D Node system demonstration on a Divertor RH use case and discusses the system applicability to other ITER RH systems. Parts of the 3D Node development are published in [1],[2].
[1] Niu, L., et al. "Eye-in-Hand Manipulation for Remote Handling: Experimental Setup", International Conference on Robotics and Mechatronics, 2017.
[2] Niu, et al. "Robust Pose Estimation with the Stereoscopic Camera in Harsh Environment ", International Symposium on Electronic Imaging, 2018.

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P2.134 Rescue tool for ITER Blanket Remote Handling System

If a remote-handling system were to become stuck and unrecoverable from the vacuum vessel, the fusion reactor would be forced to cease all operations. Proven recovery technology must be established for remote-handling systems of fusion reactors to ensure the system is recoverable from expected failures. The recovery technology for the ITER Blanket Remote Handling System is in the form of rescue scenarios, in which a rescue tool is introduced into the vacuum vessel and then remotely positioned and secured to the failed joint of the remote-handling system where it is then actuated externally. To this end, the rescue tool must be capable of being remotely positioned from every conceivable angle. This paper presents the design and test results of the rescue tool of the ITER Blanket Remote Handling System.
We designed a rescue tool having both guide and compliant mechanisms for six degrees of freedom (6DoF); motors, reduction gears, and a hexagonal socket interface to drive failed joints; and image processing markers for remote positioning. For compliance testing, the rescue tool was mounted to a mock-up of a failed joint, both with and without the compliant mechanism enabled, and the reaction forces during mounting for both cases were compared. For integration testing, the rescue tool was positioned using a robot vision system having a hand-eye camera, and then mounted to the mock-up. We found that the compliant mechanism greatly reduced the reaction force and allowed for remote positioning and installation of the rescue tool and conclude that the rescue tool developed can be used in rescue scenarios for ITER.

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P2.136 Rationale for the selection of the operating temperature for the DEMO vacuum-vessel

In magnetically confined fusion devices the plasma operation takes place in a hermetically sealed vacuum vessel (VV) of unconventional size and shape that enables the crucial high-vacuum environment.
Apart from this basic purpose the DEMO VV has to fulfil several additional requirements.
It has to provide support to the in-vessel components (IVCs) in all operational conditions in particular during plasma disruptions. Due to the double-walled construction with large water-cooling channels in the inter-space it is also designed for the removal of the heat generated during plasma operation and in accidental scenarios involving failure of IVC cooling loops. The VV is also the main constituent to the neutron and radiation shielding function protecting other components in particular the superconducting coils and all maintenance areas. Therefore wall irritation and damage by neutrons need to be considered. In addition the usage of the hydrogen isotope tritium as part of the fusion fuel requires the vessel to act as the primary confinement barrier for radiation. Tritium surface retention and bulk permeation contribute significantly to the tritium inventory.

The temperature for conditioning or operation of conventional vacuum chambers is generally chosen to provide excellent vacuum conditions. Due to the additional functions of the DEMO VV the arguments for the temperature selection are multifaceted. E.g. the higher saturation pressure for higher coolant water temperatures and the relative thermal expansion between VV and IVCs must be considered, too. In DEMO also the impact of the VV coolant temperature to the overall plant efficiency is a notable factor.

This paper analyses the topic and provides rationales for choosing a suitable operating temperature for the DEMO vacuum vessel.

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P2.137 Development of welding scheme for ITER Divertor Dome

The Russian Federation is responsible for manufacturing and delivery 54 main and 4 spare Dome, that is the part of ITER divertor cassette.
The main elements of the Dome are: Umbrella manifold, Outer and Inner manifolds made of 316L(N)-IG steel, the tubes of 19,05 mm and 141.3 mm diameter with 1.65 mm and 9.53 mm of thick walls respectively made of 316L steel and 34 plasma-facing units of Umbrella, Inner and Outer Particle Reflector Plates (PFUs)
Welding of Dome involves: the laser welding of manifold plugs and PFUs covers, multi-pass orbital welding of the structural pipes with manifolds and single-pass of the connecting pipes of PFUs.
The welding of Dome is one of the most complicated and main manufacturing operations for the Dome. The complexity results from abundant of welding on Dome where the each weld shrinks and causes strong welding stresses and strains. This might cause strong warping distortions and displacement of parts of the Dome those making it impossible to obtain the dimensions specified in the drawings.
To reduce and to control the welding strains, the special jig for assembly and welding of structural pipes between manifolds was developed and the orbital welding sequence was chosen on the basis of the results of numerical simulation.
The welding shrinkage of connection pipes of PFUs has been estimated with the aim to reduce and to control the welding shrinkage and strains between manifolds stubs and PFUs and the special sequence of assembling and welding of connection tubes between manifolds and PFUs was developed.
The report highlights the results of modeling and prototyping of the above mentioned assembly schemes with the aim to obtain the correct geometry according to the drawing, methods for solving of the emerging technological problems.

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P2.142 Structural assessment based on welding distortion simulation of Vacuum Vessel PS1 Jig for in-process control

The ITER vacuum vessel (VV) is a torus-shaped, double-wall structure with shielding and cooling water between the shells. Low distortion welding techniques are chosen in order to manufacture the 4 poloidal segments (PS) composing each sector, weld them together and then assemble on site the nine toroidal sectors to form the complete torus.
Control of the distortions during the welding process is the main technological challenges since very stringent tolerances must be fulfilled at the end of the process. Finite element analysis (FEA) tools have been developed in order to predict the mechanical behaviour of the assembly (component to be welded + supporting structure) during welding. These tools allowed on one side to optimize the welding process and on the other side to design ad-hoc supporting structure (Jigs) needed to control the distortion caused by welding and to react to the huge forces generated during the process.
The jigs are the key components during the manufacturing since their failure could jeopardize the whole process.
In this paper a novel approach is proposed to design and assess the mechanical behaviour of the Jigs and to mitigate the risk related to their failure by implementing in-process control strategies based on the coupling between finite element analysis and metrology surveys/direct measurements.
This approach has been applied on the ITER VV PS1 Jig. Experimental test have been carried out in order to validate the FE model feeding the computational tool by metrology measurements. Finally an optimize configuration of direct measurements devices (instrumented washers, strain gauges, displacement sensors) has been proposed in order to live control the behaviour of the Jig during the manufacturing process by directly computing via FEA (feeding the analysis with the measurements) all the relevant mechanical variables needed to assess the structural integrity of the supporting structure.

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P2.145 Use of dimensional variation models for the PS2 Upper Subassembly of the ITER Vacuum Vessel

The ITER Vacuum Vessel is a torus-shaped, double shell stainless steel structure made up of nine welded sectors, with five being manufactured by the European Domestic Agency of the ITER Project (Fusion for Energy). Despite the large sector dimensions (around 11 x 7 x 6 metres) and the considerable weld lengths (in excess of 1.7 kilometres per sector) each sector must achieve very tight dimensional requirements, of the order of 0.1% of overall sector dimensions.

With the aim of carrying out an early assessment of the risk of dimensional nonconformities and a prompt definition of mitigation actions, a number of 3D statistical dimensional variation models have been developed by Fusion for Energy using 3DCS software. Each model is representative of the functional tolerances and assembly processes of a given sector subassembly, allowing both to predict the expected dimensional variations in specific areas and to help identify the tolerances and process accuracies contributing the most to a given variation before manufacturing activities start.

This paper describes the 3D model developed to assess the compliance with the requirements in terms of steps and gaps before welding of the first manufacturing stages of the Poloidal Segment 2 (PS2) Upper Subassembly. The document presents the statistical results obtained and the outcome of the process of replacing the statistical tolerance variations by deviations based on as-built information of the parts. Very good agreement between model results and observed deviations was found, especially when using as-built data of the parts.

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P2.148 Development of bore welding tools for ITER blanket remote maintenance

Remote welding of cooling water pipes is one of the technological challenges for maintenance of nuclear fusion reactor. In ITER, more than 1,000 in-vessel welds are performed for the installation of First Wall (FW) and Shield Block (SB), of which failures during D-T operation require complete remote handling of these pipes due to irradiation environment in the vacuum vessel. The welds in FW and SB require two separate bore welding tools, i.e. (a) FW pipe welding tool and (b) SB coaxial connector welding tool, for different bore diameter of 42.7 mm and 100 mm, respectively. This paper presents development of these in-vessel bore welding tools.
In order to accommodate with possible positional error of FW and SB, FW pipe welding tool is integrated with the weld groove alignment tool for achieving precise groove alignment before weld operation. In addition to successful welding test of prototype welding tool, the performance test of pipe alignment tool is reported, by which gap and lateral misalignment between pipes were aligned within 0.3 mm, satisfying required precision to achieve good weld quality.
The SB coaxial connector welding tool is not required to adapt with misalignment between weld grooves due to SB's guiding features. On the other hand, the SB's geometry requires this tool to go through 80 mm diameter opening and to expand to 100 mm diameter for welding operation. This tool is integrated with visual observation tool with infrared ray camera for visual inspection of welds. The design of newly manufactured prototype tool and performance test results are presented.

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P2.152 Design and structural analysis of ITER thermal shield under transportation environment

Thermal Shield (TS) in ITER tokamak reduces heat loads from vacuum vessel and cryostat to superconducting magnet structure. Its delivery is scheduled to begin from September 2018 including TS Main Components (TSMC) and manifold pipes. Unlike to other components in ITER tokamak, most of TSMC have slender structure with panel thickness of 20 mm. Due to its structural uniqueness, TSMC cannot be secured on packing box directly since forces caused by lashing ropes may induce undesirable deformation and structural damage during transportation environment. Therefore, transportation jigs are required to support and fasten TSMC. These jigs are secured to packing box instead of TSMC. In this paper, structural design and safety assessment of TSMC transportation jigs are presented. The scope of this study will be focused on the Vacuum Vessel TS (VVTS) and Lower Cryostat TS (LCTS) firstly based on the procurement plan. Finite element analysis has been carried out to estimate the safety of transportation jigs. In the structural analysis, following loadings are considered for assembled TSMC and jig: 1) Road transportation, 2) Sea transportation considering shipping route from Korea to ITER site, 3) Lifting with crane and 4) Dynamic effect of lifting crane and slings. Using the results of finite element analysis, following items are checked: a) Deformation of TSMC and jig, b) Structural safety of jig frame and c) Structural safety of lifting lugs and connections including bolts and joints. According to these analyses, several transportation jigs and their joint connections are reinforced in order to satisfy safety regulations. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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P2.153 Investigation on the in-situation pipe bending tool for the sector sub-assembly of ITER thermal shield

The main components of the ITER tokamak are assembled from the nine sub-assemblies of the 40° sectors. During the sub-assembly, the Toroidal Field coils (TFCs) are installed outside of the Vacuum Vessel Thermal Shield (VVTS) by rotation. The clearance between TFCs and VVTS is very small, in relation to the cooling tube end points, which connect to the thermal shield manifolds. The dimensional analysis results reveal that in-situ bending of the VVTS cooling tube is required, instead of pre-bending like the reference design. This paper describes the development of the in-situ bending tool and mock-up validation. The specialized suitable tool is designed and the in-situ bending procedure is verified by a mock-up test.

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P2.155 Issues of the Vertical Blanket Segment Architecture in DEMO: current progress and resolution strategies

Due to the limited irradiation lifetime of the structural material used for in-vessel components in DEMO, and subsequent future fusion power plants, it will be necessary to replace all breeding blankets within the given planned maintenance window in order to meet DEMO availability targets. It is assumed that failure of in-vessel components cannot be excluded, whilst in-situ repair is unrealistic. Hence the replacement of individual breeder blankets must be technically feasible. As such, remote maintenance replacement of the breeding blankets is a mission critical operation. The baseline concept utilises vertical segment architecture to aid in the removal of the blankets. This choice impacts on the tokamak and plant architecture and also affects operational maintenance strategy.
Within the EUROfusion PPPT program efforts have been made to perform cross work package investigations on eight Key Integration Issues needed to show the feasibility of the DEMO pre-concept design. Key Design Issue 4 is an investigation into the feasibility of the Vertical Segment Architecture blanket feasibility.
The present work documents the approach, current progress and developments within this investigation. This includes the strategy, identified risks and proposed solutions. An alternative variant,

an Equatorial Divided Blanket Segment is proposed. A design overview and the impacts, both positive and challenging, on the design of the affected systems are provided and evaluation criteria are proposed.
The removal of breeding blankets and replacement with new or refurbished components is a complex machine operation. It will require the interfacing and developing of multiple systems and components. Due to the performance trade-off between the operational performance of in-vessel components and the remote handling suitability, the interrelationships and possible interacting challenges of extracting breeder blankets needs to be consider at the pre-concept design stage.

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P2.231 Experimental activities for in-box LOCA of WCLL BB in LIFUS5/Mod3 facility

The new experimental facility LIFUS5/Mod3 has been designed, manufactured and installed to investigate the phenomena connected with the thermodynamic and chemical interaction between lithium-lead and water in case of in-box LOCA (Loss of Coolant Accident) of the WCLL breeding blanket concept and to validate the chemical model implemented in SIMMER code for fusion application. In order to fulfill these objectives, the necessary step is to obtain data, suitable to code validation, by means of an experimental campaign in LIFUS5/Mod3 facility, executed with controlled initial and boundary conditions. Specific instrumentation and dedicated data acquisition system are installed to provide meaningful and reliable data.
The experimental campaign is carried out in WCLL blanket module relevant conditions. A predefined amount of water is injected into the reaction tank at a pressure of 155 bar and different temperatures, accordingly with a selected test matrix. The initial liquid metal temperature is fixed at 330 ℃. The experimental data and results of the executed tests (i.e. pressures, temperatures, amount of injected water, and hydrogen production quantification) are reported and discussed.
The final aim of the LIFUS5/Mod3 campaign is the SIMMER code validation, applying the standard methodology to post-test analyses. Besides, the expected outcomes of the tests are the improvement of the knowledge of physical behavior and of understanding of the phenomena, the investigation of the dynamic effects of energy release towards the structures, and of the chemical reaction with the consequent hydrogen production, the enlargement of the database for code validation.

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Progress on an Ion Cyclotron Range of Frequency System for DEMO

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An Ion Cyclotron Range of Frequency (ICRF) system is one of the options to provide power on DEMO for a number of tasks, which have all been experimentally verified on present machines: breakdown assist, heating the plasma, controlling sawteeth, removing central impurities, current ramp down assist and wall conditioning. ICRF has a number of advantages for a machine like DEMO. The system has a high plug-to-power efficiency and most of the components external to the machine are sturdy, with industrial steady state capability.
A critical aspect is the need for the antenna to be in-vessel and close to the plasma. While its low power density (compared to other heating systems) is a bonus in terms of reliability, it requires as conventional in-port antennas too much valuable in-port space. Therefore, travelling wave type antennas have been proposed [1]. They can be integrated in the blanket (several ways of doing this are being considered) and use only a limited number of feeders. Initial calculations for the antenna only, indicated that the effect on the tritium breeding ratio is small. These calculations are being

refined and will include the effect of the antenna feeders. The k// spectrum is very peaked and the dominant k// value can be reduced to optimize coupling and bulk absorption. The coupling can be further enhanced with gas puffing near the antenna. Calculations of the antenna resistance show that 50 MW can be coupled with a voltage level of the order of 15 kV. The paper will discuss the proposed concepts, options to integrate the antenna in the blanket and possible remote maintenance schemes.

[1] R.Ragona et al., "The physics of the Traveling Wave Antenna (TWA) an innovative ion-cyclotron resonance heating (ICRH) system for the reactor with proof of principle demonstration on WEST ", this conference

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Multidimensional linear model for disruption prediction in JET

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The implementation of Machine Learning (ML) techniques has considerably improved the prediction of disruptions. However, they usually provide outcomes difficult to understand from a physics point of view due to their mathematical formulation. The objective of this work is to attain an interpretable equation of an accurate ML disruption predictor. The equation could be used for real-time prediction and possibly for the off-line analysis of the disruption cause. To create the linear model, in addition to physic quantities, time derivatives (TDs) have been considered. TDs represent the difference of the amplitude values of a signal X at two different times divided by their temporal difference (i.e. dX/dt). An initial dataset of 11 signals, together with 7 TDs of each quantity, have been used. To select the best combination of signals and TDs that provide the best linear model, genetic algorithms have been applied. The dataset to train the predictor includes 100 disruptive and 1000 non disruptive JET discharges. The results, obtained with an independent test database of 133 disruptive and 1330 non disruptive shots, show 97% of success rate (93,3% with at least 10 ms of warning time) and 4.3% of false alarms. These rates were achieved with a linear equation derived from the training process. It relates 8 plasma signals and TDs: 1) poloidal beta (Pb); 2) line integrated plasma density (ne); 3) plasma elongation (ELO); 4) mode lock amplitude (ML); 5) plasma vertical centroid position (PVP); 6) total radiated power (Rad); 7) toroidal magnetic field (Bt); 8) time derivative of the stored diamagnetic energy (W_der).

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Updated design of the water cooled breeder blanket for CFETR

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The Chinese Fusion Engineering Testing Reactor (CFETR) will be operated in two phases. Phase I focuses on Pfusion=200 MW, Qplasma=1−5, TBR>1.0. Phase II emphasizes DEMO validation, which means Qplasma>10, Pfusion>1 GW (e.g., 1.5GW). CFETR has updated its core parameters in 2018. The major /minor radius have been changed from R=5.7m/a=1.8m to R=7.2m/a=2.2m. It is also required that one blanket design can cover operation of both phases of CFETR. However, fusion power in Phase II is 5~7.5 times larger than that in Phase I, which brings great challenges.

To meet CFETR Phase I and II operation conditions at the same time, a new water-cooled ceramic breeder (WCCB) blanket scheme is proposed in ASIPP, based on making trade-offs in the design considering TBR, tritium release temperature in breeder, and heat removal capability of coolant. The new scheme still employs the mixed breeder pebble bed of Li2TiO3 and Be12Ti, RAFM steel as structural material, and tungsten as armor material of the first wall. Pressurized water of 15.5MPa

is chosen as coolant with 285oC inlet/325oC outlet. The unique feature of this design is that it has two independent cooling routes. When CFETR operates at the lower power for Phase-I, only coolant circulating system 1 is employed and coolant circulating system 2 is set idle with no feeding coolant. This can increase operation temperature of tritium breeder and consequently benefit the tritium release process. When CFETR operates at the higher power for Phase-II, both cooling systems are put into use to enhance the heat transfer of the coolant and ensure that the temperature of blanket materials stay lower than the allowable limits.
In this contribution, the updated WCCB blanket design is presented corresponding to the latest core parameters and requirements of CFETR. Its feasibility is evaluated and validated from the aspects of neutronics and thermo-hydraulics.

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SPIDER Integrated Commissioning

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The construction of SPIDER, the prototype of the full-size, radio frequency, negative ion source of the ITER
Heating Neutral Beam Injectors, has been completed at the Neutral Beam Test Facility in Padova (Italy). All
SPIDER components have been delivered and have successfully undergone the site acceptance tests.
The mechanical components (vessel, ion source, accelerator, and calorimeter), power supply systems,
auxiliary systems (gas and vacuum and cooling), basic diagnostics (thermocouples and imaging), control and
interlock systems are being prepared for the operation by means of a step-by-step procedure, referred to
as integrated commissioning, aiming to progressively achieve the coordinated interoperation of all SPIDER
plant systems.
The execution of the integrated commissioning requires a well-defined organization covering personnel
safety, shift-based operation, definition of roles and their interaction.
The path to achieve full integration comprises a sequence of activities including:
- One-to-one plant system integration with the control and central interlock systems;
- Functional operation of power supply systems with the control and central interlock systems;
- Insulation and thermal tests of the transmission line and power supply systems;
- Functional operation of the auxiliary systems with the control and central interlock systems;
- Functional integrated operation of power supply and cooling systems;
- Performance tests.

The paper, after briefly summarizing the status of the SPIDER experiment, will initially focus on the organization of the integrated commissioning. The execution of the commissioning campaigns will then be
presented along with the technical and scientific issues encountered and the solutions applied to solve
problems. Finally, the paper will discuss the transition from the SPIDER integrated commissioning to the
SPIDER operation.

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Manufacturing, installation, commissioning and first results with LTCC-3D high-frequency magnetic sensors on TCV

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Innovative high-frequency magnetic sensors have been designed and manufactured in-house for installation on the Tokamak à Configuration Variable (TCV), and are currently routinely operational during the TCV experimental campaigns. These sensors combines the Low Temperature Co-fired Ceramic (LTCC) and the classical thick-film technologies, and are in various aspects similar to the large majority of the magnetic sensors currently foreseen for ITER (410 out of 454 are LTCC-1D).
The TCV LTCC-3D magnetic sensors provide measurements in the frequency range from 1kHz to 1MHz of the 3D perturbations to the parallel, namely quasi-toroidal (deltaBPAR), poloidal (deltaBPOL) and normal (to the flux surfaces, deltaBNOR) magnetic field components. The design principles were aimed at increasing the effective area and self-resonance frequency of the sensor in each measurement axis, while reducing the mutual and parasitic coupling between them. The physics requirements are set by the installation of two high-power / high-energy Neutral Beam Injection systems on TCV, namely studying fast ions physics, coherent instabilities and turbulence in the Alfvén frequency range.
We report on the manufacturing, installation and commissioning work for these high-frequency LTCC-3D magnetic sensors, and conclude with an overview of the first experimental results obtained with this system. We show that the LTCC-3D data for deltaBPOL are in agreement with those obtained with the standard Mirnov sensors installed on TCV in the frequency range where the respective data acquisition overlap, routinely up to 125kHz and up to 250kHz in some discharges. Moreover, the LTCC-3D data provide new insights on the deltaBPOL fluctuations in the higher frequency range from 125kHz/250kHz to 1MHz, on the deltaBPAR component, which is not available using the TCV Mirnov sensors, and on the deltaBNOR component, which is available using the TCV vessel-mounted saddle loops but only up to ~3kHz (due to the wall penetration time).

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Recent progress in developing a feasible and integrated conceptual design of the WCLL BB in EUROfusion Project

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The Water-cooled lithium-lead breeding blanket is in the pre-conceptual design phase. It is a candidate option for European DEMO nuclear fusion reactor. This breeding blanket concept relies on the liquid lithium-lead as breeder-multiplier, pressurized water as coolant and EUROFER as structural material. Current design is based on DEMO 2017 specifications. Two separate water systems are in charge of cooling the first wall and the breeding zone: thermos-dynamic cycle is 295-328℃ at 15.5 MPa. The breeder enters and exits from the breeding zone at 330℃. Cornerstones of the design were the single module segment approach needed to avoid that He gas bubbles formed from the Li are accumulated in the breeding zone close to the first wall; a water manifold between the breeding blanket box and the BSS with the twofold function to cool the structures of the inboard segments and to maximize the neutron shielding. The breeding blanket segment is attached with the vacuum vessel by means of a plate with a thickness of 100mm. This is in charge to withstands the loads due to normal operation and selected postulated initiating events. Rationale and progresses of the design are presented and substantiated by engineering evaluations and analyses. Water and lithium lead manifolds are designed and integrated with the two consistent primary heat transport systems, based on a reliable pressurized water reactor operating experience, and six lithium lead systems (i.e. three connected with the inboard and three connected with the outboard segments). Open issues, areas of research and development needs are finally pointed out.

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Full coverage infrared thermography diagnostic for WEST machine protection

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The WEST platform aims at testing ITER like W divertor targets in an integrated tokamak environment. To operate long plasma discharge, the IR thermography is required to monitor the main plasma facing components by means of real time surface temperature measurements, while providing essential data for various physics studies.
To monitor the new divertor targets, the WEST IR thermography protection system has been deeply renewed, to match with the new tokamak configuration. It consists of 7 endoscopes located in upper ports viewing the whole lower divertor and the 5 heating devices. Electronics devices and computers allow a real time data processing at a frame rate of 50 Hz, to ensure the protection of the main plasma facing components during plasma discharges by a feedback control of the heating devices injected power, and the data storage of ≈3 Gb/s IR images.
Each endoscope provides 2 views covering 2 divertor sectors of 30° (toroidally) and 1 view of a heating antenna. Each optical line is composed of a tight entrance window followed by a head objective which forms an image transported through the endoscope by a series of 4 optical relays and mirrors, up to a camera objective. Finally, 12 IR cameras specially developed for WEST environment capture the thermographic data, at the wavelength of 3.9 µm, with a 640x512 pixels frame size.
This paper provides a comprehensive description of the design options and diagnostic technologies: optics, mechanics, electronics, hard & software, cameras, as well as the assembling and environmental constraints. The laboratory characterization procedures are explained (Modulation Transfer Function MTF, slit response, calibration), and the measurement performance results are given (spatial and thermal resolution, temperature threshold). Finally, first results obtained during experimental campaign in WEST are presented and analyzed.

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An enhanced near-term HCPB configuration as driver blanket for the EU DEMO

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The Helium Cooled Pebble Bed (HCPB) breeding blanket is the reference blanket proposed for the EU-DEMO. A concept for the HCPB based on a cooling plate (CP) "sandwich" architecture built in Multi-Module Segments was developed for the EU-DEMO 2015 tokamak baseline. This architecture significantly improved the tritium breeding performance with respect to the former "beer-box"-like concept and keeps the circulating power at a moderate level (≈130 MW), improving also former figures (>200 MW). However, the still questionable technology readiness of a helium-cooled based Primary Heat Transfer System (PHTS) with such circulating power and the need for a simpler architecture to improve the blanket reliability still reveals critical deficiencies of the "sandwich" concept. This has led to a design revision towards establishing an enhanced, simpler, near-term configuration. Such configuration is based on a fission-like hexagonal arrangement of radial fuel-breeder pin assemblies built in Single Module Segments. This concept has been implemented in the latest EU DEMO 2017 baseline and features advanced ceramic breeder (mixed Li4SiO4 + Li2TiO3) for enhanced pebble bed thermo-mechanics and Be12Ti neutron multiplier for improved safety, minimizing the tritium retention, swelling and reactivity issues when compared to pure Be. The paper reports a basic successful set of neutronic, thermo-hydraulic and thermo-mechanical performances, with a manufacturing and assembly strategy, highlighting the simplicity of this new configuration and its integration in the PHTS. This new HCPB architecture has indeed an especially key positive impact on the Balance of Plant, which it leads, together with other enhancements in the PHTS, to a remarkably low circulating power (70÷80 MW), enabling the use of mature, state-of-the-art He turbomachinery and solving the question about the low technology readiness. The paper concludes with a discussion on the future development steps and about the revised set of requirements for the EU HCPB TBM and associated testing.

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New trends of gyrotron development at KIT: An overview on recent investigations

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Electron-Cyclotron-Resonance Heating and Current Drive (ECRH&CD) is one of the preferable auxiliary heating systems for magnetically confined nuclear fusion plasmas. ECRH&CD is widely used in tokamaks and helical machines for start-up and plasma control.
Since many years KIT is strongly involved in the development of high power gyrotrons for use in ECRH. KIT is pursuing two development lines: (i) the conventional, hollow cavity gyrotron and (ii) the coaxial cavity gyrotron. In the frequency range up to 170 GHz the conventional 1 MW gyrotron is the state-of-the-art for CW operation.
KIT is also pushing conventional cavity gyrotrons from 1 MW to 1.5 MW in a common project with IPP Greifswald. In this contribution we will report on first investigations towards a 1.5 MW, 140 GHz gyrotron.
Coaxial cavity technology has the advantage of higher power capability, in particular at higher frequency.
A short-pulse modular 170 GHz, 2 MW coaxial cavity gyrotron is being upgraded to allow pulse extension up to approximately 100 ms and up to 1 s in a second step. For this purpose the cooling performance of the main sub-components such as beam tunnel, cavity, coaxial insert and launcher has been developed and substantially improved. The design of the magnetron injection gun (MIG) has been significantly revised. In parallel an inverse MIG has been designed and manufactured. Both MIG designs satisfy the criteria for the suppression of the electron trapping mechanisms.
For a future DEMOnstration fusion power plant two challenging trends with respect to gyrotron features are recognized: (a) the operating frequency will be above 200 GHz and (b) the requested total efficiency of the gyrotron should be as high as possible. KIT is addressing these requirements by investigating both technologies for its performance at a frequency well above 200 GHz and we started careful analysis of multi-staged-depressed collectors.

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Combining research with safety: the Wendelstein 7-X video diagnostic system

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A ten-channel overview video diagnostic system was installed and commissioned at Wendelstein 7-X (W7-X) optimised stellarator. The cameras serve both for surveillance of the first wall (ensuring safe device operation) and allow for physics studies. The wide range of applications is ensured by a highly flexible data acquisition and on-board data processing. Combination with magnetic field line simulations even allow to reveal 3D structures which are relevant to the development of island divertor scenarios on W7-X.
In the second operation phase of W7-X, OP1.2a, eight channels were equipped with EDICAMs, 1.3 Mpixel intelligent CMOS cameras and two Photron (SA5 and SX-2) 1.0 Mpixel fast CMOS cameras, observing the visible radiation emitted by the plasma. The fast cameras can be equipped with interference filters as well; most of the measurements were done using a C-III filter.
The supervision of the complete torus interior (typically at 100 Hz) is provided by the EDICAMs, featuring non-destructive region-of-interest (ROI) readout capability; this feature allows the monitoring of smaller areas in parallel to the full frame overview at much higher rates (up to 10 kHz). The small ROIs can be used, among others, to study hot-spot evolution (e.g. strike lines on the divertor). Video data taken by the EDICAMs is streamed through a powerful FPGA board, suitable for real-time image processing. The user can define events,

such as light intensity thresholds to provide automatically provide trigger signals for the plasma control system or other diagnostics, including the EDICAM system itself.
Extremely fast phenomena in the 10 microsecond timescale are observed with Photron cameras, revealing 3D the rotation of filamentary structures. These capabilities make the cameras a diagnostics for 3D SOL transport providing new insights in the turbulence of the SOL plasma.

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Hydrogen Adsorption Performance for Large-scale Cryogenic Molecular Sieve Bed

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Cryogenic hydrogen adsorption using molecular sieve beds is considered to be one of the main candidate processes for recovery of produced tritium from purge gas in breeding blankets and it has been chosen for separation of hydrogen isotopes in Tritium Extraction System (TES) of Korean Helium Cooled Ceramic Reflector (HCCR) Test Blanket System (TBS). Various absorbents and their performance have been studied for the cryogenic adsorption using small-scale experiments, however large-scale experiments comparable to TBS-relevant scale are required to have sufficient confidence for component design and performance prediction of Cryogenic Molecular Sieve Bed (CMSB) for the TBS and beyond. To properly evaluate hydrogen adsorption performance for large size CMSB, a series of experiments have been performed using PGLoop facility constructed and operated in National Fusion Research Institute. The experimental conditions were set to include breeding blankets relevant parameters. The hydrogen partial pressure ranges from 100 Pa to 600 Pa, and testing with different swamping ratios, flow rates and total pressures were conducted to see their effect on the performance. While a slight reduction in hydrogen adsorption performance is observed compared to the small isotherm experiments, which can be attributed to size effects, it is shown that the experimental results reasonably agree well with existing literature data.

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New capabilities and upgrade path for the DIII-D neutral beam heating system

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The Neutral Beam Injection System (NBI) on the DIII-D tokamak includes eight ion sources operating nominally at 75-80 kV, each capable of injecting up to 2.5 MW for plasma heating and current drive. As DIII-D physics experiments evolve to explore new regimes in fusion energy research, the capabilities of the NBI systems are being improved to help provide the necessary tools.
One of the NBI projects well underway is the systematic upgrade of all beamline internal components and high voltage systems, allowing an increase in the injected power as well as an extension of the pulse length for all beam systems. The ability to smoothly vary beam energy and perveance has also recently been developed and a new algorithm that combines variable beam energy (VBE) and variable beam perveance (VBP) has been written and successfully tested. In addition, an extensive new project is underway to modify one beamline to inject off-axis beams in either the co- or counter-plasma current direction, adding significant flexibility to the system. Also, the MACE device (Miniature Arc Chamber Experiment) has recently been constructed and commissioned, allowing research into several issues that affect ion source operations and specifically targeting an improvement in reliability during helium beam operations. Finally, exploration of the edge physics

in tokamaks has benefited from a new capability to inject very low energy beams (15-20 kV), driving current at the edge of the plasma.
A description of these new capabilities and some results from experiments will be presented here.

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Manufacturing experience and commissioning of Large Size UHV Class Vacuum Vessel for Indian Test Facility (INTF) for Neutral Beams

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Indian Test Facility (INTF) is designed for "Full characterization of the Diagnostic Neutral Beam (DNB)" for ITER, to unveil the possible challenges in production, neutralization and transportation of neutral beam over the path length of ~20.67m. This facility consist of a vacuum vessel (with volume >180m3) which has been designed and manufactured as per the rules of ASME Sec.VIII Div. 1, to house and provide an ultra-high vacuum (UHV) environment for DNB operational components i.e Beam Source, Beam Line Components, High Voltage Bushing.

As per functional requirements, INTF vessel is fabricated from AISI 304L, in cylindrical shape (4.5mD,9mL), with the unique attribute of 'detachable top lid' to allow access for internal components during installation and maintenance. As per the best of authors' knowledge, it is the biggest UHV vessel with this configuration realized ever, it was therefore essential to establish a systematic approach when moving from the 'non-conventional design' to 'non-conventional manufacturing'. During this manufacturing, top lid is cut from the shell itself, which demands controlling the deflection, arising due to stress relaxation caused by welding and shell rolling. This deflection has been controlled by installing the specially designed fixture. Further, distortion monitoring during the welding of large flanges was carried out and following to that machining parameters was controlled to achieve the flatness ~1.2mm over the vacuum sealing periphery of 9mx5m for achieving leak rate 10-9 mbarl/sec.

Following fabrication, vacuum level of 8E-06 was demonstrated with local and global leak rates of 1E-09mbarl/sec and 1E-07mbarl/sec respectively. This paper presents the experience and methodologies generated in the establishing the manufacturing protocols to achieve the distortion control, deflection requirements and vacuum demonstration for large size UHV vessel with detachable top lid configuration. The learnings during this manufacturing are expected to be useful for ongoing and upcoming equipment with similar challenges.

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Standardized integration of ITER diagnostics equatorial port plugs

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The Diagnostic Shielding Modules (DSM) are secondary containers of diagnostic Port Plugs where shielding and diagnostics components have to be integrated. For the development of equatorial DSMs several key requirements have to be met. The total dry Plug weight shall not exceed the allowable maximum (45 t) in other to guarantee the consistency of the design with the specification of other interfacing systems. A shielding capability able to limit Shut-Down Dose Rate in the Interspace Zone (man access corridors) below 100 μSv/h is required as to allow the human accessibility

needed for the sealing disassembly as well as for the inspection and maintenance of diagnostics in the interspace. Remote Handling compatibility for refurbishment of the Port Plug and maintenance of all systems and components integrated inside is necessary as well. These operations involve activated components that are processed in the Hot Cell.
The Port Integration team has developed the Modular Equatorial DSM able to answer the weight/shielding/RH requirements following the ALARA principle while offering standard infrastructure for the common integration problems.
This paper describes the key aspects considered in the development of the Modular DSM, a concept that offers a structural and shielding schemes capable to meet SDDR limits within stipulated weight limits being compatible with the principle of assembly and testing, meaning that once installation and setting-up is complete, a system should not be taken apart for final implementation of other systems. This fairly simplifies the tenants design constraints and integration process adding flexibility. In this regard the Modular DSM may be understood as a diagnostics friendly concept since it follows a component, subsystem and system design, implementation and testing approach with certification at each assembly stage. The DSM solution is currently the integration approach followed for First Plasma Port Plugs which have already faced the Preliminary Design Review.

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Recent achievements of the Pd-Ag membrane technologies in tritium extraction system applications

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The main expertise of the Membrane Laboratory of ENEA Frascati is related to the study and development of Pd-based membrane technologies (both permeators and catalytic reactors), which are one of the reference processes in the fuel cycle of nuclear fusion reactors. Principal characteristics of Pd-based membranes are infinitive hydrogen selectivity, elevated hydrogen permeability, modularity, reduced cost and low energy consumption.
In the last few years, the ENEA laboratory has realized two experimental facilities for testing single and multi-tube Pd-Ag membrane modules. Several experimental and simulation activities have been carried out to evaluate the application of these technologies in the tritium extraction system of the solid blanket concept. In addition an important work has been done to optimize the membrane modules design.
This paper presents most significant results obtained in the two facilities during the He-H2 permeation and heavy water decontamination tests under several operating conditions. During the permeation tests, many He-H2 feed flow ratios have been investigated giving the possibility to experimentally describe the behavior of the Pd-Ag membrane module under DEMO-relevant conditions. Water decontamination tests have been performed by exploring the effect of different catalysts and reactions (i.e. isotopic swamping and water gas shift). All the results are discussed to define the scenario in which Pd-Ag membrane technologies can be considered an effective solution for the tritium extraction system. For the identified scenario a preliminary dimensioning of the system is also presented.

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The CNESM neutron imaging diagnostic for SPIDER beam source

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The ITER neutral beam test facility under construction in Padova will host two experimental devices: SPIDER, a 100 kV negative H/D RF source, and MITICA, a full scale, 1MeV deuterium beam injec-

tor. A detection system called Close-contact Neutron Emission Surface Mapping (CNESM) is under development with the aim to resolve the horizontal beam intensity profile in MITICA and one of the eight beamlet groups in SPIDER, with a spatial resolution of 3 and 5 cm2 respectively. This is achieved by the evaluation of the map of the neutron emission due to interaction of the deuterium beam with the deuterons implanted in the beam dump surface. CNESM uses nGEM detectors, i.e. GEM detectors equipped with a cathode that also serves as neutron-proton converter foil. The diagnostic will be placed right behind the SPIDER and MITICA beam dump, i.e. in an UHV environment, but the nGEM detectors need to operate at atmospheric pressure, so to contain the detector a vacuum sealed box has been designed to be installed inside the vacuum vessel and at atmospheric pressure inside. The box design was driven by the need to minimize the neutron attenuation and the distance between the beam dump surface and the detector active area. This paper presents the status of the CNSESM diagnostics. It describes the detector box and the different phases followed during the installation of the diagnostic on the SPIDER beam dump. Also the general layout of the diagnostic as part of the SPIDER experiment will be discussed. Finally the preliminary design of MITICA CNESM diagnostic will be introduced. This work was set up in collaboration and financial support of Fusion for Energy.

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Magnetically insulated baffled probe for measurements in the edge plasma of tokamaks

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A magnetically insulated baffled (MIB) probe offers the advantages of direct measurements of the plasma parameters (including plasma potential, electron temperature and ion temperature etc.), while being non-emitting and electrically floating[1].The MIB probe was constructed by retracting the conducting plug of a classical Langmuir probe inside an insulating tube placed perpendicular to the magnetic field lines. The retracting distance of the collector inside the ceramic tube was estimated assuming classical and anomalous mechanisms of the electron cross-field diffusion and taking into account particles losses inside the tube. The results of the MIB probe measurements in a HMHX (High Magnetic field Helicon experiment) (typical plasma parameters: electron density >1013 cm-3 for Ar helicon plasma discharge at magnetic field up to 6300 G) are presented. The MIB probe designs proposed for edge diagnostics will increase the capability to characterize separately plasma properties in real-time for understanding of underlying physics in the edge plasma of tokamaks (EAST).

Keywords: helicon-wave excited plasma, plasma potential, MIB probe

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Updated design and integration of the ancillary circuits for the European Test Blanket Systems

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The validation of the key technologies relevant for a DEMO Breeding Blanket is one of the main objectives of the design and operation of the Test Blanket Systems (TBS) in ITER. . In compliance with the main features and technical requirements of the parent breeding blanket concepts, the European TBM Project is developing the HCLL (Helium Cooled Lithium Lead) and HCPB (Helium Cooled Pebble Bed)-TBS, focusing in this phase on the design life cycle and on R&D activities in support of the design.
In this context, a sound design of the TBS ancillary systems is mandatory for the reliable operation of the two TBS in ITER. This is the basis for the success of the TBM Project, with the full scientific exploitation of the experimental results and return on experience in view of the DEMO Breeding Blanket development. .
The TBS ancillary systems are mainly circuits devoted to the removal of thermal power and to the extraction and recovery of the tritium generated in the Test Blanket Modules. They are:
• The Helium Cooling System (HCS);
• The Coolant Purification System (CPS);
• The Tritium Extraction System (HCLL-TRS, HCPB-TES);
• The Lead Lithium Loop.
Their conceptual design was deeply analyzed during the ITER Conceptual Design Review in 2015. The present design takes into account the main CDR recommendations as well as the implementation of the requirements related to ITER operation, safety principles and physical space constrains. The CPSs were redesigned to fulfill the requirements of tritium concentration in the coolant, and CPSs and HCSs were moved from CVCS to TCWS area with integration activities very complex to comply with ITER physical interfaces.
Finally yet importantly, the preliminary design and integration of TAS (Tritium Accountancy Station), which implements the fundamental function of the tritium accountancy for TBS, is presented and discussed.

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A new generation of power supplies for pulsed loads

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Pulsed power supply systems are employed in many fusion projects and in other scientific applications. A novel power supply was specifically developed to feed pulsed loads (low duty cycle), as the resistive or superconducting coils used to produce high magnetic fields for some seconds or longer. Thanks to the integrated energy storage devices, it is not necessary to draw directly from the electrical grid the high pulsed power required for the load. In fact, a huge amount of energy (more than 800 kJ and 220 Wh) can be stored at low power (even through a single-phase 10 A plug, as normally used for household appliance) inside an easily movable cabinet. This energy is promptly available at higher power when required. A significant fraction of the energy delivered to the load can be recovered for successive operations.
Without such energy storage, all the power supply devices must be oversized. Therefore, the related activities can be performed only in locations provided with adequate power. Moreover, a specific solution was implemented to achieve a very fast energization of the magnets.
The maximum output current and voltage of the power supply are 2 kA and 300 V (600 kW), respectively. The current ripple is kept within ±10 A at the flat-top and the acoustic noise is minimized by the converter's high switching frequency.
The energy storage capability is intrinsically scalable: the setup can be rearranged and the storable energy can be updated even after the installation (for example, to increase the pulse duration).
The first power supply unit was installed to feed the external coils of the compact spherical tokamak PROTO-SPHERA, located in the ENEA laboratories in Frascati. This unit can replace a previous system with comparable performances, but this system occupies an extremely larger volume and needs a 20-kV connection at the input.

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VERDI detector benchmark experiment at the ENEA 14 MeV Frascati Neutron Generator

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For future fusion plants, such as DEMO, there is a great need for detectors capable to accurately monitor neutrons under the harsh conditions imposed by fusion environment. In particular, detectors required in Test Blanket Modules must be capable to accurately measure neutron fluence under high and variable neutron count rates, high gamma background, high temperature and high and variable magnetic fields. The Novel Neutron Detector for Fusion (VERDI) project aims to develop a detector which will provide a robust approach for neutron detection in fusion plants. The detector comprises a composite low activation matrix compound capsule containing a defined concentration of added metallic elements. The neutron fluence and energy spectrum will be inferred by analysis of the multiple gamma lines produced by the activation of the metallic elements. The key innovation lies in the use of the composite ceramic capsule which is capable to withstand the extreme environment of a fusion plant. In this work, the candidate metallic elements for the VERDI detector are defined. FISPACT-II radionuclide inventory code was used in order to simulate irradiations under fusion relevant neutron spectra and calculate induced activities and their evolution in time. A benchmark experiment was performed at the ENEA Frascati Neutron Generator to demonstrate the feasibility of VERDI detectors to measure neutron fluence under a reference fusion-relevant field. A number of prototype VERDI detectors was fabricated, tested out of field and irradiated under DT neutrons (14 MeV). Gamma-ray measurements were performed aiming in the detection of both short-lived and long-lived product isotopes. The results in terms of realized reactions and induced activities were compared to the respective calculated data using the FISPACT-II code and a very good agreement was observed. In the next level of development, VERDI detectors will be tested under real fusion conditions at JET Long Term Irradiation System.

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Development of Pb-16Li technologies for DEMO Reactor

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Three of the four breeder blanket concepts for a DEMO Reactor use the eutectic Pb–16Li enriched at 90% in 6Li as breeder material: Helium Cooled Lithium Lead (HCLL), Water Cooled Lithium Lead (WCLL) and Dual Coolant Lithium Lead (DCLL). Moreover the WCLL is one of the blanket concepts that will be qualified in the ITER reactor, therefore the development and design of lead lithium loops and auxiliary systems is essential. The main functional requirements that Pb-16Li systems have to fulfill are:
• To circulate the liquid Pb-16Li through the blanket;
• To extract the Tritium produced inside the breeder modules from Pb-16Li (this function is shared with the Tritium Extraction System);
• To control Pb-16Li chemistry and to remove accumulated impurities;
• To avoid tritium permeation into primary coolant.
In the framework of EUROfusion Consortium, the design of the Pb-16Li loops for the breeding blankets and R&D activities are planned. The present work aims to describe the activities performed in order to achieve the following objectives: i) design and integration of the Pb-16Li loops inside tokamak building, ii) development and characterization of antipermeation and anticorrosion coatings in Pb-16Li, iii) development and design of tritium extraction system, iv) design chemistry control

system for Pb-16Li loops, v) perform MHD analyses taking into account also the impact on tritium transport in breeding blankets and perform safety analysis of water/Pb-16Li interaction due to LOCA inside the WCLL blanket.

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SPIDER in the roadmap of the ITER Neutral Beams

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To reach fusion conditions and control plasma configuration in ITER, the next step towards establishing nuclear fusion as viable energy source, suitable combination of additional heating and current drive systems is necessary. Among them, two Neutral Beam Injectors (NBI) will provide 33MW hydrogen/deuterium particles electrostatically accelerated to 1MeV; efficient gas-cell neutralisation at such beam energy requires negative ions, obtained by caesium-catalysed surface conversion of hydrogen/deuterium atoms in the ion source. ITER NBI requirements have never been simultaneously attained; so a Neutral Beam Test Facility (NBTF) was set up at Consorzio RFX (Italy). Experiments will verify continuous NBI operation for one hour, under stringent requirements for beam divergence (<7mrad) and aiming (within 2mrad). To study and optimise NBI performances, the NBTF includes two experiments: MITICA, full-scale NBI prototype with 1MeV particle energy; SPIDER, with 100keV particle energy, aiming at testing and optimising full-scale ion source. SPIDER will focus on source uniformity (~1.5m2 beam area), negative ion current density and beam optics. The SPIDER experiment, just entered into operation, will profit from strong numerical activities, simulating experimental scenarios, and refined diagnostic instruments, providing thorough plasma and beam characterisation.
The talk outlines the worldwide effort towards ITER NBI realisation. Coupling of radiofrequency to plasma, voltage holding with magnetic fields, distribution of caesium emitted from the evaporators will be addressed by SPIDER experimentation, supplemented by other small facilities specifically devoted to these issues. This talk will also outline the studies carried out in the ELISE facility (IPP-Garching, Germany), equipped with a half-size source, and in NIO1 at Consorzio RFX, which are supporting ITER NBI development. Concerning beam optics physics, specific issues are also addressed by joint experiments at QST and NIFS (Japan). Finally the talk will present the results of the first experiments in SPIDER, including the preliminary source plasma characterisation.

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A smart architecture for the DEMO fuel cycle

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In the framework of the EUROfusion DEMO Programme the EU has elaborated a completely novel and most innovative fuel cycle architecture, driven by the need to reduce the tritium inventory to an absolute minimum.
To achieve this goal, batchwise processes used in the fusion fuel cycle so far were replaced by continuous processes wherever possible. This includes the change from discontinuous cryopumping to mercury based continuous vacuum pumping with practically zero demand on cryoplant power, and the introduction of thermal cycling ab- and adsorption processes for isotope rebalancing in the tritium plant instead of large cryogenic distillation columns with tritiated liquid hold-ups. To further reduce inventory, the well-known approach to route all exhaust gas through the tritium plant has been abandoned in favor of a three-loop architecture. There, superpermeable metal foils are introduced in the divertor ports to separate a pure DT stream which is then immediately recycled to feed the pellet injection systems. To increase the core fueling efficiency, optimization potentials in the design of the high field side pellet injection systems are being exploited. Finally, a unified

fuel cycle simulator is under development on a commercial software platform in order to identify optimization potentials within the fuel cycle, to allow impact studies, and on a long term to support the development of tailored control and operational strategies.
The talk will present the first integrated and consolidated design point of the fuel cycle based on the 2017 European DEMO baseline. It is shown how the DEMO requirements are picked up and affect system level performance. Examples are given for integration issues and how they were solved (remote handling, divertor integration, plasma control). Finally, a roadmap is delineated which illustrates the remaining R&D efforts needed to achieve at a validated and complete conceptual design until the mid 2020s.

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High Temperature Superconductors for Future Fusion Magnets and Industrial High Current Applications

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High-temperature superconductors (HTS) have the potential to enable the operation of a future fusion reactor at higher magnetic fields (> 14 T) or at higher temperatures compared to conventional low- temperature superconductors. In particular, the operation at high magnetic fields with good temperature margin is perceived to be an important advantage of HTS in a fusion power plant.
Fusion magnets require high current superconductors, which are embedded in a stainless-steel jacket for mechanical support against Lorentz forces and actively cooled by a forced flow of coolant. Design challenges and cable proposals for winding packs based on HTS conductors will be presented.
HTS CrossConductor (HTS CroCo) is a high current HTS conductor with high current density to be fabricated in long lengths. The recent progress of HTS CroCo fabrication will be shown and a toroidal field coil winding pack design based on such HTS CroCos will be presented as an example to demonstrate the principle feasibility of HTS for future fusion magnets.
The knowledge gained in the design of high-current conductors for fusion magnets and the experience to fabricate HTS CroCo strands in different geometries enable the design and construction of HTS DC cables for industrial high current applications as well. Key design aspects and promising fields of applications of HTS high current cables beyond fusion magnets will be presented.

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P3.007 Fusion device components Hot Helium Leak Test facilities main requirement and lessons learned from the WEST project

Actively Cooled Plasma Facing Components (ACPFC) are required to allow for long plasma discharges in magnetic fusion devices. Prior to their installation, the integrity of ACPFC has to be checked under relevant experimental conditions in order to prevent serious water leaks in the vacuum vessel.

Since 1990, the French Magnetic Fusion Research Institute (CEA/IRFM) has developed specific leak test procedures adapted to the different component types and operational constraints. Meanwhile several facilities have been built to perform Hot Helium Leak Tests (HHLT).

To ensure a high enough quality leak test, these facilities must meet certain criteria such as low helium residual value and high sensitivity in the test bed vessel (~5.10-11 Pa.m3/s). Specific technics are used to check the component under representative experimental conditions (i.e. pressurization up to 5 MPa, temperature range from 20 to 250℃) when carrying out the CEA IRFM special leak test procedures.

This paper describes the equipment that were developed to perform the HHTL and the different procedures applied. A summary of the results of the tests carried out on approximately 500 components during the WEST assembly period is also presented. The lessons learned from these tests led to some modifications of the procedures, in particular with respect to the number of thermal cycles, pressurization levels and cycling, and heating steps. The goal is to perform an adapted test ensuring a reliable result in as a short time as possible.

Some further optimizations of the test duration are eventually suggested, which could have a very important impact on the overall assembly schedule of fusion devices under construction, like in particular ITER.

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P3.028 Design of KSTAR ECH launcher for long pulse high performance Tokamak operation

Since KSTAR first plasma operation, ECH played a key role in obtaining various experimental results such as ECH preionization, ECH-assisted startup, plasma rotation study, impurity transport study, high poloidal beta operation, and long pulse operation. The main heating systems in KSTAR are NBI and ECH which are planed to provide 12 MW NBI by 2019 and 6 MW ECH by 2020 to prepare long pulse, high beta discharges. The ECH launcher should also be able to operate in steady state and to serve as central/off-axis heating and current drive, q-profile control, NTM, and sawtooth controls. The launcher has features such as continuous operation by water cooling and realtime precise-high speed beam control by connection with Plasma Control System (PCS) for supporting above experiments, in particular for NTM suppression. The whole control system with PCS has control loop rate of 1 kHz and a latency of about 10 msec at the mirror speed of 40 degree/sec. In addition, highly localized power deposition is very beneficial to the experiments, so the curvature and size of the mirrors in the launcher have been optimized to minimize the beam diameter in the resonance range. In this paper, the design of ECH launcher and its control system are presented with the test results.

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P3.029 Test of polarizers for the 105 GHz ECRH system on J-TEXT

To match the electron cyclotron wave with the plasma efficiently, we design two polarizers including a linear polarizer and an elliptical polarizer for the 105 GHz electron cycltron resonance heating system on J-TEXT. The linear polarizer is mainly used to change the rotation angle of the wave, while the ellipticity of the wave is regulated by the ellptical polarizer. The sinusoidal grooves are applied to improve power capability. The poalrizers were manufactured, and we set up a low power test platform to measure polarization characteristics of the polarizers. To improve accuracy of measurement, the detecor was calibrated with the standard wave source. The test reuslts agree well with the design parameters. Therefore, we think the desigend polairers can meet requirements of the electron cyclotron resonance heating system on J-TEXT.

P3 / 1079

P3.032 Research on establishment methods of basic survey control network

for HL-2M tokamak

Due to the high precision requirements of HL-2M tokamak sub-components assembly, the survey control network with high precision should be established. With high accuracy distance measurement of laser tracker system and distance intersection method, the local coordinates of reference points(or the relative locations of the reference points) in the survey control network are calculated. There are several steps in the simulation: 1) the ideal coordinate of reference points and laser tracker stations generation in the ideal global coordinate system. 2) the ideal distance calculation between reference points and stations and an error of distance measurement with a normal distribution added(sigma=2.5 and 5 micron). 3) the local coordinate(in the stations) calculation with distance intersection method and coordinate transformation from the local coordinate to the ideal. 4) the performance evaluation(delta: root-mean-square error of distance between the calculation and the ideal in the global coordinate system). The evaluation shows that delta is about 15-25 micron with sigma=5 micron and will be less than 10 micron with sigma=2.5 micron.

P3 / 1109

P3.062 High throughput spectrometer for plasma ion temperature fluctuations measurement on EAST

A high throughput spectrometer has been developed for the measurement of the plasma ion temperature fluctuations on EAST tokamak. The designed spectrometer operates at the spectral range of 527.5±5 nm, then the emission lines from CVI at 529.1nm and NeX at 524.9nm can be observed simultaneously. The collimated and focus lenses are specially developed in order to realize the maximize transmission and keep the optimized image quality. One customized transmission grating sandwiched by two identical BK7 prisms from Wasatch Photonics is used. The clear size of the grating is 205×120 mm and the grating has very high diffraction efficiency of >75% for unpolarized light. One existing high speed Phantom V710 camera (1280×800 pixels, active area: 25.6×16 mm) is used to test the performance characteristics of the spectrometer at the first step. In this paper, an overview of the spectrometer will be shown and the laboratory tests of the spectrometer performance are presented.

P3 / 1265

P3.218 MIRA: a high fidelity system/design code for advanced fusion reactor system analysis

Fusion systems codes are essential computational tools aimed to simulate the physics and the engineering features of a fusion power station. The main objective of a system code is to find one (or more) reactor configurations, which simultaneously comply with the physics operational limits, the engineering constraints and the net electric output requirements.
As such simulation tools need to scope many design solutions over a large parameter phase space, they rely on rather basic physics and engineering models (mostly at zero or one-dimensional level) and on a relatively large number of input specifications. With reference to the ongoing EU-DEMO conceptual design workflow, the systems codes are interfaced to the detailed transport codes and engineering platforms, which operate in much larger time scales. As a consequence, a design feedback to the system codes is unlikely to take place fast and efficiently.
In light of such a challenging frame of reference, a new high fidelity fusion system/design code, referred to as Modular Integrated Reactor Analysis (MIRA), has been recently developed at Karlsruhe Institute of Technology. MIRA incorporates, into a unique computing environment, a mathematic algorithm for the utmost tokamak fusion problems, including: (i) 2D free-boundary magnetic equilibrium, (ii) neutron, photon and charged particles poloidal wall loadings, (iii) radial neutron/gamma

transport in core reactor components and (iv) full 3D engineering characterization of the toroidal and poloidal field coils.
To elucidate the capabilities of the full package, this work comprises a glance on the architecture and functional logics of MIRA, together with an applicative example on the EU-DEMO 2015 baseline. Accordingly, the major deviations from basic zero-dimensional system analysis approach and the implications on the reactor design are illustrated. Finally, a sensitivity study on the reactor build, intended as the radial and vertical thickness of the key core reactor components, is also presented.

P3 / 1215

P3.168 Electromagnetic analysis of the water cooled breeder blanket for CFETR Phase II

China Fusion Engineering Test Reactor (CFETR), the next-step fusion device of China, is proposed to design and operate in two phases. The physical parameters and machine sizes of CFETR have been updated in 2018. It is required that one blanket design can cover two operation phases of CFETR. The water cooled ceramic breeder (WCCB) blanket for CFETR phase II, one candidate CFETR blanket option, is been updated in Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP). Electromagnetic(EM) analysis model has been built by ANSYS code. Eddy current and EM force results under different operation conditions are calculated for future structure analysis and optimization. Simultaneous, an analysis of ripple induced by reduced activation ferritic/martensitic(RAFM) steel in WCCB blanket is evaluated with the method of Static Magnetic Analysis. The method for reducing the ripple has been studying in this paper.

P3 / 1080

P3.033 Plasma equilibrium based on EC-driven current profile with toroidal rotation on QUEST

In the EC-driven (8.2 GHz) steady-state plasma on QUEST, plasma current seems to flow in the open magnetic surface in the outside of the closed magnetic surface in the low-field region according to plasma current fitting (PCF) method. First, plasma equilibrium solution was fitted assuming all plasma current is flowing in the inside of the LCFS. It was solved within isotropic pressure profile by EFIT code. Opposite-polarity current density region appeared in the high-field region. The elongation ratio of the LCFS is about unity and the aspect ratio is lower than two. Second, particle orbit-driven current density profile showed hollow current profile and shifted outward. The equilibrium fitted by SU-EFIT does not coincide with the orbit-driven current profile even with not-nested magnetic surfaces. The numerical solution of Grad-Shafranov equation could not show the steady-state current profile with skin and peaked profile. Third, considering the toroidal rotation, the equilibrium is fitted within nested magnetic surfaces by SU-EFIT. Though the plasma magnetic axis shifts outward due to the centrifugal force, the opposite polarity current does not disappear in the high-field region. The equilibrium will be fitted also in EC-driven (28 GHz) steady-state plasma and the fitted toroidal rotation speed will be compared with the measured one.

P3 / 1147

P3.100 Replacement of the NSTX-U Inner PF Coils

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The National Spherical Torus eXperiment Upgrade (NSTX-U) is an experimental device funded by the U.S. Department of Energy (DOE) at the Princeton Plasma Physics Laboratory (PPPL). NSTX-U (http://nstx-u.pppl.gov/home) is an upgrade of the original NSTX device that operated successfully for more than 10 years as a proof-of-principle demonstration of the ST concept.

During early phases of operation of NSTX-U, one of the poloidal field coils experienced a turn-to-turn fault that necessitated a complete shutdown. Extensive reviews were performed to determine the root cause. A precise cause could not be identified but several contributing factors became apparent. It has since been decided to replace all three Inner PF upper/lower coil pairs.

The coils are water-cooled copper solenoids that are pulsed up to 20kA for durations ~ 2 seconds, repeated every 1200 seconds, 20,000 times over the lifetime of NSTX-U. During pulses the conductors experience a temperature rise that is nearly adiabatic, and hoop stress due to their self-field and the background field from neighboring coils. Between pulses, cold water enters the inlet and a cooling wave propagates through the coils as slugs of cold water heat up to the conductor temperature and then pass through the coil to the outlet.

New designs have been developed for the Inner poloidal field (PF) coils that factor in findings from the root cause analysis. Detailed analysis has been performed for both the energized state and the cooldown cycle.

Prototype coils are being fabricated to qualify multiple suppliers so parallel production lines can be deployed to minimize the overall fabrication time.

This paper reports on the replacement of the Inner PF coils including lessons learned from the failure, improvements in design, results of analysis, experience with prototype coils, and progress with the fabrication of replacement coils.

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P3.035 Data model implementation in ITER data archiving system

ITER's CODAC archiving system manages currently three different sets of data: DAN, SDN and PON, that correspond with the data that is transmitted by different networks: Data Archiving Network (data produced by data acquisition, diagnostics and data analysis), Synchronous Data Network (real time control network), and Plan Operational Network (control data). In this sense, ITER's CODAC data archiving system needs to manage a wide variety of types of data. In some cases, the structure of this information is very simple and can be defined with a primitive type, such as float, double or integer, but in many other cases type structures are much complex, and data model has to manage multidimensionality with dynamic dimension sizes, metadata embedded types with header and footer sections, or user defined composition of types.
ITER is a big project that involves many different developers and integrators. Data model technology must be flexible enough to cover almost all their necessities. Additionally, one of the main requirements for ITER arching system is data accessibility. Archived data must support long-term archiving independently on ITER systems evolution, and backward data compatibility must be guaranteed. This important requirement mostly relays in a robust and standard data model implementation that avoids opaque sections.
The aim of this work is to present the data model in ITER's CODAC data archiving system. It includes detailed description of the design and main clues of its implementation in all its phases: data acquisition, data archiving, data retrieval and user presentation.

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P3.038 Assessment of linear disruption predictors using JT-60U data

Recent research on disruption prediction shows that predictors based on analysing the amplitude evolution of magnetic signals outperforms the results obtained by using simple thresholds. To accomplish this, the disruptive and non-disruptive information of discharges can be compressed into two centroids in a particular parameter space (PS). During a running discharge, points in the PS are assumed to show a disruptive behaviour if they are closer to the disruptive centroid. By using a single signal, the plasma behaviour is evaluated with a simple linear equation in two variables (the ones that form the PS), which allows a straightforward real-time implementation. This new class of predictor can be used for next devices such as JT-60SA or ITER. To test the method, data obtained in the JT-60U high-beta operation have been used (in particular the magnetic perturbation time derivative signal). The PS is a 2-dimensional space formed by consecutive amplitudes of the above signal. 76 disruptive discharges and 78 non-disruptive ones have been used in two different approaches. Firstly, the centroids have been determined with 40% of the available shots (disruptive and non-disruptive). The preliminary results with the remaining discharges show 98.2% of success rate, 5.2% of false alarms and 17 ms of average warning time. A second approach has been the development of an adaptive predictor from scratch in which the discharges are taken in chronological order and the predictor learns in an adaptive way. The first predictor utilizes 1 disruptive discharge and 2 non-disruptive discharges. New computation of centroids is carried out after each missed alarm. With this adaptive predictor the success rate is 97.3% (only two re-trainings were necessary), the false alarm rate is 19.7% and, on average, the warning time is 59 ms. Details, interpretations and potential application to JT-60SA will be discussed.

P3 / 1089

P3.042 Remote control of a high-speed pellet injector and data synching & sharing tools

The four-barrel, two-stage gun Ignitor Pellet Injector (IPI) was developed in collaboration between ENEA and ORNL to provide cryogenic Deuterium pellets of different mass and speed to be launched into tokamak plasmas with arbitrary timing. The prototype injector is presently located at Oak Ridge (TN, USA), and is normally operated locally through a control and data acquisition system developed in LabVIEW, handling multiple subsystems: vacuum, cryogenics, propulsion valves and pellet diagnostics. More recently, a remote-control system has been set up, based on RealVNC®, which allows to fully operate the IPI from a control room in Italy. Tools for data transfer and storage into ENEA ICT area have also been provided1. A Staging Storage Sharing system, named E3S, previously developed over the ENEA ICT infrastructure, using OwnCloud as architectural component, is used for file syncing and sharing of the IPI data. It provides a homogeneous platform able to store and share heterogeneous data produced by many data acquisition systems in large nuclear fusion experiments like the tokamaks. The cloud storage technology has allowed to design an architecture based on concepts such as: i) data integrity and security, ii) scalability, iii) reliability. The deployment of E3S works on the pellet facility and allows storing data acquired by the diagnostic systems onto a wide area distributed file-system for sharing, as well as for remote data access based on MDS+ tool, and integrated with MySQL metadata accessible by means of web-services. The paper presents the E3S and a performance analysis of the architectural components. The performance analysis has been carried out with customized benchmark tools on a test bed consisting of a HPC cluster over Infiniband mounting a high performance storage.

P3 / 1071

P3.024 Anode power supply for the electron cyclotron resonacne heating system on J-TEXT tokamak

Based on the Gycom gyrotron of a diode type with a single-stage depressed collector, a 105GHz/500kW/1s electron cyclotron resonrance heating sytem is being developed on J-TEXT tokamak. To modulate output power of the gyrotron, we designed a 33kV/1A anode power supply based on the pulse step modulation technology. The power supply consists of 40 modules with output voltage of 800 V and 10 modules with output voltage of 100 V. In this way, the output voltage of the anode power supply can be adjusted accurately without pulse width modulation. We tested the andoe power supply with a dummy load. The test results indicated that the designed power supply can meet requirements of the ECRH system on J-TEXT.

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P3.090 Manufacturing status of ITER PF6 double pancakes

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The Poloidal Field (PF) coils are one of the main sub-systems of the ITER magnets. The Fusion for energy (F4E) is in charge of supplying 5 Poloidal field coils (PF2-PF6) as in-kind contributions to ITER project. In 2013, F4E commissioned the task of PF6 coil fabrication to Institute of Plasma Physics Chinese Academy of Sciences (ASIPP). The PF6 coil consists of 9 double pancakes (DPs). Before starting production of the DPs, Small- and full-scale trials are being performed to demonstrate and optimize fabrication procedures, as well as proving the stability of the fabrication tooling system. From 2017 the first DP for production was launched. So far, 7 of 9 DPs have been wound and 4 of 9 DPs impregnated. Measurement accuracy of 0.05% was applied in conductor forwarding control and ±0.5mm radial build-up for each turn were achieved, which demonstrated good winding performance. DPs were fully impregnated and the flatness of 1.5mm and profile of ±3mm were obtained, which passed AC/DC high voltage test afterwards. The final position of DP terminations after the joint boxes installation were within the tolerance of 3mm. Above results reflects good progress on PF6 DP fabrication at ASIPP premises .

P3 / 1094

P3.047 First Wall and Divertor Protection in JET-ILW: Assessment of Reliability after fifty Hours of Plasma Operation

The risk of damaging the metallic PFCs on JET-ILW by beryllium melting or cracking of tungsten owing to thermal fatigue requires a reliable active protection system: it shall avoid damage to the plasma-facing components (PFCs). To address this issue, a real-time protection system comprising newly installed imaging diagnostics, real-time algorithms for hot spot detection and alarm handling strategy has been implemented into the JET protection system and successfully demonstrated its reliability. It must be active in every plasma discharge and shall become especially important during the execution of the experiments in the coming D-T campaign, where stationary plasmas with an additional power of 40MW for 5s but tolerable wall heat loads and impurity concentration are required.
The real-time protection system has been operated routinely over 12000 discharges (>50hours of plasma operation) since 2011. Within the last experimental campaigns 2-3% of the plasma discharges were successfully terminated by this system, avoiding material overheating and damage. The different hot spot detection algorithms fulfil their tasks with a high reliability: triggering of false alarms

is less than in 0.5% of all plasma discharges. The real-time protection system is an essential tool for JET operation and the experience gained can contribute important ideas and methods to the design of the ITER plasma control system.
Future development of the JET real-time first wall protection is focused on the D-T campaign and on operation near ITER relevant conditions. D-T operation at JET will cause failure of camera electronics within the Torus hall due to significant increase of the hard radiation level (neutrons/gammas). To provide the reliable wall protection needed during coming D-T campaign, two camera systems, equipped with new optical relays to take the images and the cameras outside of the biological shield, have been installed on JET-ILW and calibrated with an in-vessel calibration light source.

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P3.051 Measurement of neutron flux emitted from 14 MeV DT generator

The paper reports on measurements of neutron flux emitted from a 14 MeV DT neutron generator. Such devices are widely used in material sciences, industry, medicine, etc. The used neutron generator (NSD-35) provides a controllable emission of a stabilized neutron flux, up to about 2·E8 neutrons per second in 4pi angle. According to manufacturer, more than 90% of the neutrons emitted are 14-MeV neutrons.
The detection techniques we used include the activation method and polyallyl-diglycol-carbonate (PADC) track detector. Activation samples, after irradiation and cooling, were measured using scintillation probes, e.g. LaBr(Ce). During experiment detectors were placed in azimuthal plane, at various positions around generator, and irradiated for chosen time.
In order to record neutrons using the track detector, we used special converters which, after neutron irradiation, emitted charged particles. The converters made of several materials were proposed and tested, both numerically and experimentally. Part of the detector was shielded by a thin metal filter, in order to extend energy range of the detectable particles. We also performed optimization studies of the converter and filter sets in various configurations. The particles were afterwards registered by the track detector. After developing, tracks were observed, measured and counted using an optical microscope. Precise calibration of the used detector made possible to properly identify energy, range and type of the recorded particles. Monte Carlo simulations allowed us to compute number of charged particles emitted from the converter, which then could be registered by the used detector.
The MC calculations were also used for analysis and interpretation of the measurement results. Neutron flux calculated on the basis of the described methods was in good agreement with the specification of the neutron generator.

P3 / 1138

P3.091 Preliminary design of the island divertor coils on J-TEXT tokamak

The high heat load on divertor target plate is one of the essential issues for future fusion reactors. In stellarator, the island divertor configuration has a long magnetic field line connection length. It is beneficial to increase the equivalent radial transport and the power decay length, and consequently reduce the peaking heat load on the divertor target plate. Therefore, it is significant to explore the possibility of applying this configuration to tokamak. This work uses the external resonant magnetic perturbation (RMP) to generate the boundary magnetic island on J-TEXT, and then studies the feasibility of forming the island divertor configuration on tokamak.
In previous studies of dynamic ergodic divertor (DED) on TEXTOR, it has been observed that magnetic islands and ergodization form at the plasma boundary, which can increase radial transport of particles and distribute the heat flux to a larger area. However, they have found that the 2/1 tearing

mode is easily to be excited by the deep penetration of 3/1 base mode. In order to avoid exciting core locked mode, we select the 4/1 island. But higher m and stronger magnetic shear may increase the difficulty to form a large boundary island. To form an island with critical width of about 2.5 cm, the applied 4/1 RMP field needs to exceed 6 Gauss in the edge for the typical boundary plasma parameters on J-TEXT.
Considering to the features of J-TEXT, two types of RMP coils are designed, i.e. the saddle coils outside the vacuum vessel and the S-type modular coils in-vessel. The dominant RMP field produced by these two coils is 4/1 RMP. The width of the 4/1 magnetic island is about 2cm calculated by the NIMROD code. It is consistent with the empirical formula. The detailed design and other researches will be presented in the conference.

P3 / 1106

P3.059 Laser transfer technique using wavefront correction and homogenizers for Thomson scattering diagnostics

A Thomson scattering system providing electron temperature and density profile requires a high energetic laser. YAG lasers amplified by flash lamps (~50-100 Hz of repetition frequency with a few Joules output) sometimes suffer from wavefront distortion and peaked beam profile. The wavefront distortion deteriorates beam profile in the far field. Since a focusing lens is used toward the plasma to ensure good spatial resolution (~10mm), the deteriorated profile degrades spatial resolution and the collection optics miss measurement of scattered photons, which results in degradation of SN ratio. We employ a deformable mirror, which can deform the mirror surface using multiple piezo actuators so that the wavefront of the reflected beam becomes flat. To test the application, a YAG laser to be used for JT-60SA, which has some wavefront distortion (0.46 wavelength of peak-to-valley), is employed. Use of deformable mirror enables an improvement of the wavefront distortion to 0.1 wavelength (peak-to-valley). The expected beam waist in the JT-60SA plasma can be reduced from 13mm to 5mm, which is preferable for collection optics with limited collecting power.
A peaked beam profile causes damages on optical components such as lenses and mirrors. This limits a long beam transfer and use of amplifiers with crystals providing population inversion. We developed a new homogenizer, which is a diffraction optical element (DOE) to change the profile from peaked to non-peaked. The output at the image was designed to have a super-Gaussian shape. We use a continuous-wave laser (Gaussian) of 1064nm as an input to evaluate scale with the non-peaked profile. The homogenizer having the super-Gaussian shape sustained a non-peaked profile longer by a factor of 3.5 (=700mm/200mm) than that designed to have a top-hat shape at the image. This sustained scale is applicable for various optical components.

P3 / 1204

P3.157 Fusion neutron generation and two-dimensional neutron measurement by imaging plate detector

This research aims to develop a two-dimensional analysis of neutron flux within the blanket modules by using a compact discharge device as neutron source and imaging plates for detector. Neutron detectable imaging plate is composed photostimulated luminescence (PSL) material and converter such as gadolinium, allowing a high spatial resolution neutron radiography in a wide dynamic range of 10^5 with high sensitivity even at the neutron production rate of 10^6 n/s level, when exposed for a several hours. Previous researches showed that feasibility of a quantitative measurement by the neutron imaging plate as PSL intensity measured in a read out process is proportional to number of neutrons on the converter. In the experiment, deutrium–deutrium fusion neutron (2.45 MeV) is generated from a discharge-type fusion neutron source and then blanket samples are irradiated to measure the neutron transport. Neutron production rate from the source was enhanced by coating

Ti thin film on the cathode. Li targets (Li2CO3 powder and LiPb plates) and thin Au foils were placed in front of the imaging plate. Neutron image was read out by photostimulation using red visible light. Simulation of neutron flux distribution in the experimental system was performed by Monte Carlo n-particle (MCNP-5) transport code with FENDL-2.1 library and then compared with the neutron image. Calibration of PSL into neutron flux was carried out by using 252Cf neutron source. After irradiation for approximately two hours with neutron production rate higher than 10^6 n/s, neutron image was successfully obtained. The image showed high PSL intensities in area where more neutron is irradiated, while decreased PSL intensities were observed behind the LiPb block caused by neutron capture by Li-6, in consistent with the simulation result. Neutron transport analyses measuring two-dimensional neutron distribution in a blanket module are experimentally verified.

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P3.116 Continuous laser illumination for in situ investigation of tungsten erosion under transient thermal loads

Transient heat fluxes up to 1 MJ·m-2 on divertor area are expected during operation of ITER. They can lead to severe erosion of plasma-facing components. Studies on tungsten damaging under thermal shocks are widespread, but they are mainly concentrated on postmortem analysis of the exposed samples. Main feature of the experiments conducting on electron beam based test facility called BETA is application of optical diagnostics for in situ investigation of material erosion during transient heating.
Two optical diagnostic techniques are employed on this facility: recording of thermal radiation of the target and its imaging with illumination by light of continuous laser. Fast CCD cameras and photodiodes followed by ADC are used for recording of spatial and time-resolved temperature distribution on the surface. Tungsten tiles were polished to mirror state for this research. Sample surface is illuminated with continuous laser so that specularly reflected beam passes along axis of lens of the optical system. Major crack network is clearly visible on surface images obtained by CCD camera in this configuration. A fraction of the laser radiation is deflected by beam splitter to a set of optical fibers which records intensity of specularly reflected and scattered by the surface laser beam. Tungsten tiles were exposed to heat loads below melting threshold with pulse duration up to 300 μs. Two distinct and time-separated processes were observed by continuous recording of intensity of scattered laser radiation. The first one occurs during and shortly after exposure for 50 ms. The second one has very sharp edge (<30 μs) and appears with relatively long delay (up to 1 s) when the target already cooled down to almost room temperature. The latter phenomena can be interpreted as formation of major crack network on the sample surface.
This work was supported by Russian Science Foundation (project №17-79-20203)

P3 / 1200

P3.153 Nuclear analysis for the Preliminary Design Review of the ITER Equatorial Port #12

The ITER Equatorial Port #12 is a first plasma port, which has undergone the Preliminary Design Review (PDR) in November 2017. In support of the PDR, the following nuclear analyses have been conducted: i) the nuclear heat has been calculated in the port plug, as one of the principal thermal loads considered in the design, and ii) the shutdown dose rates (SDDR) have been estimated in the port interspace, in relation to the SDDR requirement of 100 μSv/h (after 106 s of cooling time) to host planned in-situ maintenance activities.
The nuclear heat was determined in a high spatial resolution mesh of 2x2x2 cm3, covering the complete EP#12 port plug and diagnostic systems. In relation to the SDDR, the role played by each

system, both in terms of neutron flux leakage and activation, was quantified in matrices of responsibilities. This study was then complemented in the next step, where the neighbor ports were explicitly modelled and the radiation transport and activation were considered in the complete neighborhood of EP#12.
As a result, the thermal loads due to radiation were quantified for the main elements of the EP#12, to be considered in further thermo-mechanical analyses. The calculations have shown that the SDDR in the maintenance corridors of the EP#12 is 189 μSv/h and 182 μSv/h, which is well outside the requirement to host the planned maintenance activities. The role of each system was quantified, indicating that the integration of the Visible – Infrared camera could be optimized. Two radiation cross-talks have been identified: i) a cross-talk from the In-Vessel Viewing System (IVVS) port through the cryostat and giving rise to the SDDR in the right corridor of EP#12, and ii) a cross-talk from both EC-UL ports towards the EP#12 interspace. Conceptual shielding proposals to mitigate the situation were analyzed.

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P3.164 Progress of KSTAR pellet injection system

Pellet injection system of 20 Hz has been operated in KSTAR (Korea Superconducting Tokamak Advanced Research) since 2016. The pellet can be injected to the plasma with different size, velocity and frequency during plasma experiments. The pellet trajectory is interesting topic in KSTAR so the related investigation is carried out outside of tokamak at first. We introduce the preliminary result of trajectory test and the preparation work for the density feedback control with pellet diagnostics system and PCS (plasma control system) in this paper.

P3 / 1165

P3.118 Effects of divertor leg length on detached plasma formation in linear plasma device

Steady-state fusion reactors and DEMO reactors will have much higher heat flux from the core than that from ITER, which itself exhibits heat flux that is several times larger than that available in the current fusion reactors. The detached plasma is effective for reducing heat load. However, since the generation of detached plasma requires to introduce a large quantity of gas, there is concern about the backflow of neutral particles. The long-leg divertor which is one possible solution for that has been proposed, but the relationship between the divertor-leg length and the backflow suppression / the detached plasma generation is not clear. Recently, advanced divertor aiming at additional heat removal by improving the magnetic field structure has been studied. Even in these diverters, because detached plasma is used for heat remove, it is necessary to clarify the relationship between the divertor-leg length and the neutral backflow / detachment plasma. Therefore, we conducted fundamental experiments to investigate the variation of characteristics about the backflow and the detached plasma by leg length and the magnetic field. We had reported the characteristics of the neutral backflow under the curved divergent magnetic field [1]. In a recent work, we experimentally examined the relationship between the leg length and the backflow / the detached plasma generation. The experiments had been performed by a linear divertor simulator, known as the Test plasma Produced by Directed current discharge for the Sheet plasma IV (TPD-Sheet IV) [1,2]. The orifices as a divertor-leg with different length were installed in front of the target region, respectively. The plasma parameters such as the electron temperature and electron density were measured by a Langmuir probe.

[1] T. Takimoto et al., Fusion Eng. Des., 124 (2017) 235–238.
[2] A. Tonegawa et al., J. Nucl. Mater., 313-316 (2003) 1046.

P3 / 1168

P3.121 Status and improvement of the high heat flux W monoblock type target with graded interlayer for application to DEMO divertor

Operational reliability of the divertor target relies essentially on the structural integrity of the component, in particular, of the material interfaces, where thermal stresses tend to be concentrated. To improve bonding quality, a concept developed in the frame of the EUROfusion project WPDIV for the DEMO divertor, consists in the use of functionally graded material (FGM) as interlayer between armor material (W) and structural material (CuCrZr). Due to the reduced thermal path, the advantage to use FGM, with a thin thickness (~25 µm), is the mitigation of fatigue cracking of the armor surface due to recrystallization. Due to industrial process readiness, thin FGM was deposited (from 100% W to 100% Cu) by physical vapor deposition. In 2017, six manufactured mock-ups, equipped with tungsten blocks of 4 mm thickness, demonstrated their capability to withstand 25 MW/m² and thermal fatigue tests at 20 MW/m² (1000 cycles).
From the experience gained during mock-ups development, new optimized mock-ups are under manufacturing. Optimization consists in the fabrication time reduction and in the mock-up lifetime improvement. To reduce fabrication time, the number of coated blocks is reduced and tungsten blocks with 12 mm thickness (as for the ITER-divertor) are used. Moreover, in order to improve component life time with regard to coolant induced corrosion/erosion of the CuCrZr tube, the CuCrZr tube thickness is increased by 30% to 1.5 mm (as for the ITER divertor). With finite element modeling (FEM), we show that such modifications may degrade the mock-up performance under high heat flux loading. With FEM, it is shown that augmentation of CuCrZr mechanical resistance is possible using thick FGM interlayer (~350 µm). For these improved mock-ups, rationales and required manufacturing steps will be reported in this paper. Moreover based on metallographic examinations, the recrystallization state, in comparison to the expected one, will also be presented.

P3 / 1176

P3.129 Characterising the impact of castellations on the efficiency of induction heating during testing in the HIVE facility

High heat flux testing is a vital part of engineering component validation for fusion technology. The Heat by Induction to Verify Extremes (HIVE) facility is designed to improve the practicalities of this aspect of component testing. It provides a faster turnaround for smaller concepts and a more cost-effective approach by utilising induction heating within a small vacuum vessel.

Due to the potential complexity of induced current paths in an induction heating system, the amount and homogeneity of power coupled to a component is difficult to model. This uncertainty increases where components have geometrical features such as the grid pattern castellations that often feature on plasma facing components in fusion reactors.

This project investigates the influence of various castellation patterns on the coupling characteristics of HIVE. It shows for example that as the grid density of castellations is increased, the applied heat flux increases from 4.5 MW/m2 to 6.85 MW/m2 for an input power of 30kW, due to an improvement in the efficiency of the inductive coupling from 13.4% to 20.6%. Additional experimental factors affecting efficiency and homogeneity of heating are also discussed.

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P3.131 Direct numerical simulation of coolant water flow in plasma

facing component

The simulation plays an important role to estimate characteristics of cooling in plasma facing components such as blanket and divertor. An objective of this study is to perform large -scale direct numerical simulation (DNS) on heat transfer of turbulent flow on coolant water flow. The coolant flow conditions in plasma facing components are assumed to be Reynolds number of a higher order. To investigate the effect of Reynolds number on the scalar structures, the Reynolds numbers based on a friction velocity and a pipe radius were set to be $Re\tau$ = 2100 and 3150. The detailed turbulent quantities such as the mean flow, turbulent stresses, turbulent kinetic energy budget, and the turbulent statistics were obtained.

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P3.151 Thermal behavior of conceptual blankets in K-DEMO under decay heat induced by neutron irradiation

It is foreseen from the decay heat analysis that the total decay heat from the blankets reaches up to 55.6 MW immediately after two years of the full power operation of K-DEMO with the fusion power of 2.2 GW. Especially, the estimation shows that the decay heat from an outboard blanket made of Reduced Activation Ferritic Martensitic (RAFM) steel and tungsten first wall would be tens of kilowatts per module. The issue is very important for the K-DEMO operation and maintenance, particularly for the safety of the reactor. Also, the issue is very closely related to the design parameters of hot cell and component cooling strategy.
We have investigated the thermal behavior of conceptual blanket modules in K-DEMO under the decay heat by 3-D heat transfer analysis using ANSYS Fluent software. The significant decay heat just after plasma shutdown can cause thermal failure in the case of a module without any active cooling. In an outboard module, it leads to the maximum local temperature over 1300 ℃ and the temperature difference up to 840 ℃. Natural convection provides integrity to remain within the allowable temperature range of the materials used, when provided that the cool-down time is secured at least a couple of days; the overall temperature of the module is reduced to about 200 ℃ in 10 days after the shutdown. The results suggest that cool-down time of such in-vessel components should be carefully considered for the maintenance scenario and schedule. Moreover, it shows that the active cooling of the components through a primary heat transfer system is essential until the components would be sufficiently cooled down. The results would be used as a reference point for safety assessment and maintenance of K-DEMO.

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P3.15 Continuous and Cryogen-free Isotope Separation for DEMO

An important goal for DEMO is the tritium inventory reduction in the fuel cycle. For that, the residence time must be minimized and the tritium content in the individual fuel cycle sub-systems must be reduced. One activity foresees the implementation of an isotope rebalancing and protium removal unit - requires less recycling, has lower hold-up and has a lower residence time than cryogenic distillation - to process the majority of the unbalanced fuel mixture.
For this purpose, a facility is currently being developed that runs continuously and also operates under non-cryogenic conditions. The concept is to design and build a two-stage system that meets the requirements for the above mentioned process steps. The first part consists of a membrane that can meet high flow rates under continuous conditions. It is therefore economical, lowers the tritium inventory, and causes a rough separation of the hydrogen isotopes. In the second part, which aims to achieve high purity of the hydrogen isotopes, the temperature-dependent absorption and desorp-

tion behavior of the individual hydrogen isotopes in the presence of certain metals is utilized. To enhance this effect, a process is being developed that uses two alternating metal getter beds, one interacting preferably with the lighter and the other with the heavier isotope.
The facility is currently under construction with detailed design, material characterization and modelling being carried out in parallel. This paper will describe the separation principle, experimental set-up and results from modelling.

P3 / 1210

P3.163 Measurement and CFD simulation of flow mal-distribution in z-type manifolds of a fusion demo blanket

The blanket system of Korean fusion demonstration reactor (K-DEMO) has a cooling channel through which pressurized water flows to cool down the heat from nucleate heating and plasma radiation. In order to evaluate the cooling performance of blanket, a computational fluid dynamics (CFD) code has been widely used as well as used in commercial heat exchangers. However, CFD can show a large difference in the analysis results for flow distribution in the manifold depending on the geometrical type and condition of the branch pipe. In particular, in the case of K-DEMO blanket, the cooling channel has a manifold structure, in which small rectangular channels are gathered and linked with narrow flow headers, in order to prevent neutron leakage and to ensure tritium breeding ratio (TBR) by adjusting breeding materials. This design can induce flow mal-distribution and secondary flow. These phenomena are strongly dependent on turbulence model and mesh structure in CFD analysis. Therefore, in order to ensure the reliability of the blanket thermal hydraulic analysis, validation of the CFD using the experimental data is important. In this study, flow measuring experiment was performed at room conditions by fabricating the cooling channels of K-DEMO blanket. For flow measurement, PIV technique, which is one of the non-intrusive methods, was adopted to avoid any additional pressure drop in measuring channel. In addition, the CFD analysis was performed for the experiment apparatus. Specifically, three turbulence models of standard k-ε, realizable k-ε, and k-ω shear stress transport were evaluated and mesh sensitivity analysis was performed. In result, realizable k-ε and k-ω shear stress transport models showed better predictions than standard k-ε model.

P3 / 1108

P3.061 Divertor infrared thermography at COMPASS

A new fast divertor infra-red (IR) thermography system was put into operation at COMPASS. It provides full radial coverage of the bottom open divertor with pixel resolution ~ 0.6-1.1 mm/px. on the target surface (0.04-0.12 mm/px. mapped to the outer midplane) and time resolution better than 20 μs. This setup provides unique capabilities for heat flux profile measurements simultaneously in the inner and the outer divertor regions in various plasma regimes: exceptional temporal resolution secure ELM-resolved heat flux measurements, good spatial resolution is adequate for measurements of very narrow heat flux decay lengths in inter-ELM H-mode periods and for study of profile modification during the application of resonant magnetic perturbations.
Individual parts of the system are described: the fast IR camera TELOPS Fast-IR 2K, its magnetic shielding box and the positionable holder, the 1m long IR endoscope consisting of 14 Ge and Si lenses securing of-axis view from the upper inner vertical port. The special graphite divertor tile optimized for IR thermography is presented as well. It is equipped with the heating system allowing tile preheating up to 250℃, embedded thermoresistors, the calibration target (a deep narrow hole acting as a black body radiator) for in-situ system calibration including estimation of the target surface emissivity and the roof-top shaped structure increasing magnetic field incidence angles above 3 degrees.
Laboratory tests of the system performed during its commissioning are presented (system transmis-

sion, spatial resolution characterised by the slit response function) as well as examples of the first experimental results.

P3 / 1121

P3.074 The ITER radial neutron camera in-port system: nuclear analyses in support of its design development

The ITER Radial Neutron Camera (RNC) is a multichannel detection system hosted in the Equatorial Port Plug 1 (EPP 1). It is designed to measure the uncollided neutron flux from the plasma, providing information on the neutron emissivity profile and total neutron strength. The RNC structure consists of two sub-systems based on fan-shaped arrays of cylindrical collimators: the ex-port system, covering the plasma core with 2 sets of lines of sights lying on different toroidal planes, and the in-port system, enclosed in a dedicated cassette within the EPP1 diagnostic shielding module, for the measurement of neutrons generated in the plasma edge. Due to the harsh environment in which it has to operate, the design of the in-port RNC system is particularly critical both from the measurements point of view (low signal to noise ratio induced by the high level of scattered neutrons at the detector positions) and from the structural point of view.
The paper presents the results of the neutronic analyses performed with the MCNP Monte Carlo code with the aim of optimizing the in-port RNC design in order to enhance the diagnostic measurement performance and evaluating the nuclear loads that have to be withstand by its structural elements, detectors and associated components.

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P3 / 1112

P3.065 A visible light high-speed imaging system and tomographic reconstruction of plasma emissivity in J-TEXT

Visible light high-speed imaging systems (VLHIS) are widely used in imaging and diagnosing plasmas in tokamak, e.g., plasma boundary position, structures, fast fluctuations, for its visibility and easy operation. To monitor the discharge process on J-TEXT tokamak in a high temporal resolution and study of the visible light emissivity distribution, the plasma boundary shape, a new VLHIS has been constructed. The new VLHIS has a higher frame rate (up to 12000fps/s) than the old one, and the aperture on it can be changed to solve the problem of low incoming light caused by short exposure time when increasing the camera's frame rate. It has two working modes with the field of view (FOV) constant, one is for high frame rate, low resolution and another one is for high resolution, low frame rate. For the line-integrated camera images, a tangential image tomographic reconstruction technique based on camera calibration, ray tracing and solving an overdetermined set of equations is applied to compute the visible light emissivity profile of the poloidal cross section. Preliminary results about the time-averaged reconstructed plasma emissivity profile have obtained to estimate the plasma boundary shape and compared with CIII, CV radiation distribution measured by spectrograph.

P3 / 1272

P3.225 RFP based Fusion-Fission Hybrid reactor model for nuclear applications.

A hybrid fusion-fission (HFF) reactor based on a Reversed Field Pinch (RFP) configuration looks as an attractive option from both a technical (simple design, easy machine assembly and maintenance) as well as economic perspective (low investment costs due the absence of large Heating and Current Drive systems and superconductive toroidal field coils).
The hybrid reactor studied here has a RFP fusion core generating 40MWth fusion power via D-T reactions. The reactor size (R=6 m, a =1), its components design and plasma performances have been extrapolated from the results of RFX-mod, the largest RFP experiment built so far. With a plasma temperature of 9.6 keV and an ohmically driven plasma current of 20 MA, the fusion reactions produce 2.2E+19 n/s which are used to both breed tritium and induce fission reactions.
A breeding blanket is located between the superconductive central solenoid and the torus. Being composed of a lithium-lead eutectic mixture, it also acts as neutron shield.
Instead, in the configuration studied here, the circular area around the torus (16 cm width) is half dedicated to further breed tritium while the rest is occupied by the fission blanket. The three sectors (~3.5 m radial extension) containing Pu+MA (60%)-Zr (40%) rods embedded in solid lead for nuclear transmutation (keff~0.94, P~200 MWth) are spaced out by two intermediate regions (~1.75 m radial extension) that can be used for inducing fast and slow transmutations.
The results corroborate the interest for the RFP as a HFF reactor due to the simple design of the fusion reactor, the high accessibility to the blankets for maintenance and the multiple applications of the neutron flux (transmutation of plutonium, minor actinides and nuclear fission products, radiopharmaceuticals production) favoured by the blanket modularity.

P3 / 1146

P3.099 Design of the support structures of the DTT magnet system

DTT is the acronym of "Divertor Tokamak Test" facility, a project for a compact but flexible tokamak reactor which has been conceived in the framework of the European Fusion Roadmap. It will be built in Italy and shall act as a satellite experimental facility to integrate the extrapolation of the ITER results to the EU-DEMO machine. It is thus mainly aimed at the exploration of different divertor solutions for power and particles exhaust, and to study the plasma-material interaction scaled to long pulse operation. The conceptual design of the magnet system is currently under development and it will define all the characteristics for TF, CS and PF superconductive coils. The magnets must be supported in order to balance the gravitational and magnetic forces. This is achieved with five types of mechanical structures: gravity supports, pre-compression structures, inner inter-coil structures, outer inter-coil structures and centering structures. In the present work it will be reported the design choices along with the analyses and optimization studies lead to the definition of all the five types of support systems.

P3 / 1113

P3.066 Broad band ECE measuremenst in ECW heated plasmas

The Wendelstein 7-X (W7-X) experiment is equipped with an Electron Cyclotron Resonance Heating installation consisting of 10 gyrotrons capable of delivering upto 7.5 MW of Electron Cyclotron Wave power at the 140 GHz resonance in the plasma. Normally, the gyrotron power is delivered in a very narrow band of several 100 MHz around the gyrotron frequency and the gyrotrons are optimized to deliver power only at this frequency. In practice, the gyrotron central frequency will shift several

100 MHz during start-up while excitation of spurious modes inside the gyrotrons may lead to power at frequencies spaced several GHz away from the resonance. These effects do not hinder gyrotron operation, but the power at unintended frequencies is a problem for sensitive microwave receivers, typically measuring power levels in the sub μW range. For protection these generally employ notch filters such as fundamental mode interference cavities.
This paper focuses on the effect of spurious gyrotron power on the recently installed Michelson Interferometer at W7-X. In the case of a Michelson Interferometer many modes are required to enter the instrument preventing the use of single mode notch filters. More-over, the very wide band, 50 … 500 GHz means that any spurious modes from any of the 10 gyrotrons may affect the ECE spectrum. Data on spurious gyrotron modes is used to assess the potential impact on the ECE spectrum while laboratory tests on the Michelson are used to assess non-linear response of the instrument in the case of excessive power at specific frequencies. It is expected that the work is complemented with experimental data from the W7-X experimental campaign OP1.2b, scheduled to start in the summer of 2018.

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P3.227 Analysis of S-CO2 Brayton Power Cycles for Fusion power reactors

The paper focuses on the design of appropriate power cycles for fusion power reactor, two S-CO2 Brayton cycles, and its positive and negative aspects. The goal of the paper is to propose a suitable power cycle and its optimization for the European fusion power plant DEMO2. Comparison of cycles in terms of using more heat resources at once is depicted. The study gives a principal preview of main technical parameters of the suitable S-CO2 power cycles. Optimization of suggested designs in order to maximize the power of the fusion power plants is presented.

P3 / 1119

P3.072 The JET Upgraded Toroidal Alfvén Eigenmode Active Diagnostic System

The Alfvén Eigenmode Active Diagnostic system (AEAD) has undergone a major upgrade and redesign to provide a state of the art excitation and real-time detection system for JET.

The new system consists of individual 4kW amplifiers for each of the six antennas, allowing for increased current, separate excitation and real time control of relative phasing between antenna currents. The amplifiers have a frequency range of 10-1000 kHz, divided into various frequency bands by external matching filters. Due to the varying transmission line impedance throughout the frequency range, the amplifiers were designed with high resilience to reflected power with Voltage Standing Wave Ratio VSWR»10:1.

The existing signal generator and control electronics associated with amplifier control have been replaced with a digital control system incorporating a National Instruments PXI express platform and Field Programmable Gate Array (FPGA) modules for frequency, gain and phase control with a frequency and phase resolution of less than 1 kHz and 1 degree respectively. Complementing the digital control system is the Protection and Control System, which utilizes Field Programmable Analog Arrays (FPAAs) and an array of electronic devices to monitor and control the AEAD.

New capabilities such as independent antenna current/phase control, allow for improved excitation control, better definition of antenna spectrum combined with enhanced system reliability. These capabilities will be enhanced by newly installed and calibrated magnetic probes on JET allowing accurate synchronous detection of AEs. This contribution will review the new AEAD system, including legacy parts of system still in use, its unique capabilities and improvements over the previous

diagnostic system.

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P3.081 Development of the bronze processed Nb3Sn wires using various Cu-Sn-In ternary alloy matrices

The degradation of transport current property by the high mechanical strain on the practical Nb3Sn wire is serious problem to apply for the future fusion magnet operated under higher electromagnetic force environment. Therefore, increase of the mechanical strength on Nb3Sn wire is the most important research subject. Recently, we approached to the solid solution strength process on the ternary Cu-Sn-Zn matrices for the high mechanical strength bronze processed Nb3Sn wires. We found that the mechanical strength of the Nb3Sn wire was improved with increasing solid solubility of Zn as a solute element. On the other hand, there was also a trade-off relationship between the Sn and Zn composition in the bronze matrix. In order to optimize the solid solution process for the high strengthened Nb3Sn wire, a ternary bronze alloy containing Indium (In) as a solute element (Cu-Sn-In), which is thought to be more effective solute element than Zn was casted as a solid solution strengthened bronze material.
We fabricated the bronze processed Nb3Sn wires using various Cu-Sn-In matrices. In these wires, In remained into the matrices after the Nb3Sn synthesis, and then the Vickers hardness of the Cu-Sn-In matrices after the Nb3Sn synthesis heat treatment was higher than those of the conventional bronze and Cu-Sn-Zn matrices. It is suggested that indium element act more effectively as the solute element for solid solution strength process compared with zinc element and it may contribute to further mechanical strengthening of Nb3Sn wire.
In this study, effect of In element as the solute element on microstructure and superconducting properties of the bronze processed Nb3Sn wires using various Cu-Sn-In matrices was mainly investigated.
This work mainly performed to the Fusion Engineering Research Project (UFFF036-1) and the collaboration program (NIFS16KECF017) in NIFS, and also supported by the Grants-in-Aid for Scientific Research (B) (16H04621) from MEXT.

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P3.238 Heating of dense plasma to > 10 keV by femtosecond laser pulse for ICF

In the last decade, it has been intensively studied to heat a compressed DT fuel to an igniting temperatures of about 5 keV by using picosecond laser pulses. In the present work, we have investigated to create high temperature (> 10 keV) plasma at relatively high densities, by using a femtosecond laser pulse combined with a specially structured micron-sized target. The structured target is designed such that the total volume is minimized to make the resultant energy density maximum, while the inner total area is maximized to make the laser absorption maximized. As a result, the fuel plasma is heated to unprecedentedly high temperatures, because the characteristic time of plasma heating is shorter than that of hydrodynamic decompression.

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P3.027 SPIDER Gas Injection and Vacuum System: from Design to Commissioning

The ITER project requires at least two Heating Neutral Beam Injectors (NBIs), each accelerating up to 1MV a 40A beam of negative H-/D-.ions, to deliver to the plasma a total power of about 33 MW for one hour.
Since these requirements have never been experimentally met, it was recognized necessary to build-up a test facility, named PRIMA including both a full-size negative ion source (SPIDER - Source for Production of Ion of Deuterium Extracted from Rf plasma) and a full prototype of the ITER injector (MITICA - Megavolt ITER Injector & Concept Advancement). This realization is made with the main contribution of the European Union, through the Joint Undertaking for ITER (F4E), the ITER Organization and Consorzio RFX (CRFX) that hosts the Test Facility in Padova, Italy. SPIDER is a Radio Frequency ion source that has the same characteristics foreseen for the ITER NBI but with beam energy limited to 100 keV. The mission of SPIDER is to increase the understanding of the source operation and to optimize the source performance in terms of extracted current density, uniformity and pulse duration.
The paper describes the Gas injection and Vacuum System (GVS), from the analysis of requirements to the system detailed design, procurement, site acceptance tests and commissioning. In particular, it presents the rationale behind the main design choices and specific manufacturing details of the gas injection plant feeding the RF source. Furthermore, the paper describes the sensor system to measure the vacuum level, gas pressure and flow, and residual gas analysis considering the interface with the SPIDER/MITICA central interlock and safety systems. Reference is also made to safety aspects concerning the presence of H2/D2 in a closed environment as the SPIDER biological shield. Finally, the main results of the GVS commissioning with the control and interlock systems are presented.

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P3.145 Final Design and Structural Analysis of Sector Lifting Tool for ITER

Sector Lifting Tool (SLT) are purpose-built tool for the lifting and transferring ITER components. SLT consists of the Sector Lifting Tool (SLT) with the lifting attachments. The purpose of the SLT is to lift and transfer Vacuum Vessel (VV) and Toroidal Field Coil (TFC) from Upending Tool to Sector Sub-assembly Tool (SSAT). After the sub-assembly at SSAT in assembly hall, 40° Sector which is composed of VV, VVTS and two TFCs is transferred from SSAT to Tokamak in-pit. Sector Lifting Tool and the lifted components are connected with the lifting attachments. SLT and lifting attachments have to withstand component's weight with two times safety margin. The target component's weights are VV (450 tonnes), TFC (310 tonnes) and 40° Sector (1250 tonnes) during lifting by crane operation.
This paper describes the final design of Sector Lifting Tool and shows that the structural integrity has been verified through analysis. The structural analysis was performed in accordance with the ITER load specification and tooling procedure. In analysis result, the stresses and deflections turned out under the allowable ranges.

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P3.229 Implementation of the lead lithium eutectic properties in RELAP5/Mod.3.3 for nuclear fusion system applications

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P3.011 Status scientific results and technical improvements of the NBH on TCV tokamak

The TCV tokamak contributes to physics understanding in fusion reactor research by a wide set of experimental tools, like flexible shaping and high power ECRH. Plasma regimes with high pressure, a wider range of temperature ratios and significant fast-ion population are now attainable with the TCV heating system upgrade. A 1 MW, 25 keV deuterium heating neutral beam (NB) has been installed in 2015 and it was operated from 2016 in SPC-TCV domestic and EUROfusion experimental campaigns (~50/50%). The rate of failures of the beam is <5% and mostly due to arc between grids of ion optical system. The beam also stopped due to plasma density limits (to avoid excessive shine-through) or plasma disruptions.
Ion temperatures up to 3.5 keV have been achieved in ELMy H-mode, with a good agreement with ASTRA simulations adding the NB. The NB enables TCV to access ITER-like βN values (1.8) and Te/Ti ~1, allowing investigations of innovative plasma features in ITER relevant ELMy H-mode.
The advanced Tokamak route was also pursued, with stationary, fully non-inductive discharges sustained by ECCD and NBCD reaching βN~1.4-1.7. Loss channel are mostly related to charge-exchange, especially in low-collisionality plasmas as transport simulations and FIDA spectroscopy measurements confirm. Furthermore, with ECRH a new transport channel appears and might be explained by turbulent transport.
Real-time control of the NB power is planned for 2018 and will be presented together with the statistics of NB operation on the TCV. During commissioning, the NB showed unacceptable heating of the TCV duct, indicating a higher power deposition than expected. A high beam divergence has been found by dedicated measurement of 3-D beam power density distribution with an expressly designed device featuring a tungsten target on which the beam is deposited. The identification of the problem is ongoing.

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P3.022 Development of two directional beam steering ECH/CD launcher for JT-60SA

Wide range poloidal (~60°) and toroidal (~30°) beam steering capability and the reliability for high-power (0.8 MW/waveguide), long-pulse (100 s) operation are required for the launcher of the Electron Cyclotron Heating and Current Drive (ECH/CD) system in JT-60SA (Super Advanced). The two directional beam steering launcher has been designed by a linear motion antenna concept, which has an advantage of the reduction of the risk of water leakage in vacuum by aliminating flexible tubes for cooling water in vacuum.
Recently, a development of a bellows (vacuum boundary) and of a support structures for both linear and rotational motions of the steering shaft in vacuum environment has been carried out. The target lifetime of those structures is 100,000 cycles for linear motion (40 cm) and 10,000 cycles for rotational motion (30°). The cyclic tests of a full scale mock-up of the bellows structure were successful up to the target cycles without any problem. A full-scale (~7m) mock-up including four sets of the support strucutre was fabricated. It consists of a pair of anguler ball bearings made of SiN housing and balls for rotational motion and a linear guide assembled by a rail and a housing made of non-magnetic steinless steel and SiN balls. It is noted that no lubricant was used in this test. Although the rotational motion was successful up to 10,000 cycles in the test, the linear guide was damaged at 10,000 cycles, which was 1/10 of the target. In order to overcome this issue, a new linear guide with solid lubricant was developed. The linear motion of 100,000 cycles was successfully performed by the new one. Based on the mock-up test results that satisfy the requirements for the bellows and support structres, the final design of the launcher is underway targeting start of its fabrication in near future.

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P3.235 On scheduling methodology for large-scale R&D projects using TRL and critical chain and its application to fusion

Large-scale R&D projects are experiencing frequent delays due to high development uncertainties. Schedule issues are creating a series of problems that are causing delays in the entire projects by increasing the cost of projects and thereby reducing the reliability resulting in delays in timely tasks such as building the R&D facilities. In this study, considering the fact that the technology readiness level is related to the development period of large R&D projects and that it can prevent the delay by buffering the activity with high schedule risk, we developed a scheduling methodology that computes a probabilistic distribution of large-scale R&D projects, buffers the project and prevents delay and facilitates project management. In the methodology, the elemental technologies of the target system are searched and the development time, technology readiness level, and schedule risk level are evaluated by experts. Next, probabilistic delay is calculated using the Monte Carlo method. When the basic schedule is determined by taking into consideration the post-assembly relationship of the final system calculated for each activity, the difference between the basic schedule and the first schedule from which the spare time is removed is compared and the difference is stored in the buffer. In the case of a critical chain, it is put into the project buffer. In case of noncritical chain, it is set as a feeding buffer. Lastly, the calculated schedule is revised and supplemented by referring to the R&D device construction plan, which is a major constraint factor of large R&D projects. The validation of the methodology was made through comparison of simulated data and the KSTAR project data. Based on the validation result, an application for scheduling DEMO project is presented.

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P3.002 ST40: engineering commissioning first results

Spherical Tokamak (ST) path to Fusion has been proposed in [1] and experiments on STs have demonstrated feasibility of this approach. Advances in High Temperature Superconductor technology [2] allows significant increase in the Toroidal field which was found to improve confinement in STs. The combination of the high beta, which has been achieved in STs [3], and high TF that can be produced by HTS TF magnets, opens a path to lower-volume fusion reactors, in accordance with the fusion power scaling proportional to beta^2 Bt^4 V. Feasibility of low-power compact ST reactor and physics and engineering challenges of the ST path to Fusion Power will be discussed.
High field spherical tokamak ST40 (R=0.4-0.6m, R/a=1.6-1.8, Ipl=2MA, Bt=3T, k=2.5, pulse ~ 1-10sec, 2MW NBI, DD and DT operations) is now operating. TF Cu magnet in ST40 will be LN2 cooled and research is on-going on development of full-HTS magnets. TF of 3T in an ST makes ST40 to have 3 times higher field than in any operational or upgraded STs and is an important step on the ST path to Fusion.
Details of engineering design, results of commissioning, of first experiments and experimental plans will be presented. Magnetic reconstruction and visible light image of the plasma obtained using merging-compression plasma formation, as used on START/MAST [4], are in good agreement. A target for NBI has been created. We will undertake experiments on ST40 to demonstrate performance of high field ST in burning plasma regimes to support designs of next step devices on the ST path to Fusion.

[1] R Stambaugh et al, Fus. Tech. 33 (1998) 1; [2] M Gryaznevich et al, Fus. Eng. & Design 88 (2013) 1593; [3] M Gryaznevich et al, Phys Rev Lett 50 (1998) 3972 [4] M Gryaznevich, A Sykes, NF 57 (2017) 072003.

P3 / 1213

P3.166 Investigation of influence factors for oxidation reaction of trace H2 and CH4.

There are various gas components in the exhaust gas of the D-T fusion reaction. All of the hydrogen isotopes are recovered and reused as fuel, and the remaining components are released to the environment. Before releasing to the environment, all substances containing trace amounts of Q2 and Q (such as CH4) must be recovered. An oxidation / adsorption process can be used for this purpose. By adding O2 to the feed gas, the Q component is converted into Q2O by the oxidation reaction, and the generated H2O is removed by the adsorbent.
In this study, the removal efficiency of Q2 and CQ4 by oxidation reaction was investigated. H2 and CH4 were used as feed gas and Pt was used as reaction catalyst. The removal efficiencies of H2 and CH4 were measured by varying the reaction temperature, feed flow rate and concentration. And the effect of various impurities on the oxidation reaction was also investigated.

P3 / 1050

P3.003 Conceptual design of low aspect ratio superconducting tokamak with high magnetic field (T15-SC)

The next stage of the upgrade of T-15MD machine (R=1.5m, a=0.67m, B≤2T, Ipl=2MA) to the superconducting one (excluding the limits of pulse duration) with basically the same geometry is suggested. The estimations show the possibility to make a toroidal magnet with aspect ratio A=2.2, magnetic field on the axis Bo=5T, maximum magnetic field Bm= 12.5T. Such increase in Bo provides the possibility to get plasma current up to 5 MA.
The main design problem for such device is the limited space for current flow in the Toroidal Field coils and for the supporting structure of the coils. The toroidal magnet design suggests the SC-winding takes the main part of mechanical loads due to thick wall conduit that works together with coil casings. Each of 16 toroidal coil will consist of three layer: HTSC inner one, Nb3Sn middle one and NbTi outer one. Each layer will be connected with that of neighboring coils, providing three-layer toroidal winding. Each layer will be charged independently, having the same discharge time to avoid the energy exchange during charging and protection discharge.
The cooling will be done by two-phase He pushed from bottom to the top of coil by ejector pumps through the coil volume with many channels between the turns or by pushing it across the coil in horizontal direction.
This new project with its favorable combination of low aspect ratio A and really high magnetic field B is beneficial for the plasma performance and the long-pulse discharge via high fraction of bootstrap current.
The main research topics foreseen are the confinement features at high B and low A, steady-state operation and plasma wall interaction. This project will also serve as a test-stand, which is absolutely necessary for the Fusion Neutron Source SC magnet design, mastering and tests.

P3 / 1052

P3.005 A new upper divertor with internal coils for ASDEX Upgrade - status of the project

ASDEX Upgrade is a tokamak that can be operated with the strike lines in the upper and/or the lower divertor. In 2016 a project was started to develop and install a new upper divertor with internal coils and an in-vessel cryo pump. The aim is to investigate advanced magnetic configurations that may facilitate the access to detachment via an enhanced flux tube expansion and/or connection length. To realize the envisaged magnetic configurations two internal coils operated with up to 52 kAt will be installed. The conceptual design was presented in [1]. Since then critical aspects of the project

were identified and investigated in detail. These are the coil concept, the forces, in particular during disruptions, an intrinsic save power supply and last but not least, operation with co and counter magnetic field.
The coil concept and the forces are linked. Whereas the resulting forces during disruptions can be handled, a short cut due to an arc between two windings of a coil can result in excessive forces which require at least a hardening of the vessel. At present two cable designs are tested that can avoid or reduce the risk of arcing between windings. The conceptual design has shown that the two coils have to be operated anti-parallel with about 20 % imbalance in maximumo reducing the forces between the divertor- and the control coils. This will be ensured by an intrinsically save power supply. The concept of the new divertor combines the stiff coil support structure and the target support. This allows installing flat target tiles with a gap size of 1 mm and +/- 0.2 mm assembly tolerances minimizing leading edge effects. Thus, physics investigations with both directions of the toroidal magnetic field are possible.
[1] A. Herrmann, et al., Fusion Engineering and Design 123 (2017) 508-512.

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P3.006 Recent development and physical improvements of AFSI-ASCOT based synthetic neutron diagnostics at JET

New development steps of ASCOT and AFSI (ASCOT Fusion Source Integrator) based synthetic neutron diagnostics and validation at JET are reported in this contribution. Synthetic neutron diagnostics are important not only in existing tokamaks, where they are used to interpret experimental data, but also in the design of future reactors including ITER, DEMO and beyond, where neutrons are one of the few diagnostics available. Thus development and validation of realistic synthetic diagnostics is necessary for increasing confidence in existing models and future diagnostic designs.

Recent development in AFSI includes physical improvements such as implementation of plasma rotation and reduction of the fast particle contribution in thermal reactant distribution. The rotation typically changes the beam-thermal reaction rates by 1-5%, while accounting for the fast particle density consistently decreases calculated neutron rate by up to 15% depending on the discharge.

Further developments include implementation of angular dependence of differential fusion cross sections and accounting for finite Larmor radius effect, which is important for high-energy particles such as ICRH ions. Additionally, the role of data based analysis in synthetic diagnostics development with the help of comprehensive database of JET operating conditions (JETPEAK) is discussed.

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P3.008 Magnetic configurations and electromagnetic analysis of the Italian DTT device

The European roadmap to the realisation of fusion energy has identified a number of technical challenges and defined eight different missions to face them. Mission 2 'Heat- exhaust systems' addresses the challenge of reducing the heat load on the divertor targets. Divertor Tokamak Test (DTT) facility [1]-[2] has been launched to investigate alternative power exhaust solutions for DEMO. DTT should offer sufficient flexibility to be able to incorporate the best candidate divertor concept (e.g. conventional, snowflake, super-X, double null).
In this paper, the revised up-down symmetric Italian DTT device is presented. The up-down symmetrization of DTT makes possible to increase the reference values of the plasma current. At the same time, it has an impact on the costs, for which a slight revision of the main parameters has been considered, e.g. a reduction of the major radius.

Starting from a conventional Single Null configuration for the DTT facility, we investigate the feasibility and the costs of the alternative magnetic configurations on DTT. We have developed SnowFlake, X-Divertor, Super-X and Double Null configurations optimizing the plasma shape and the currents on the PF coils. The magnetic configurations feature the main characteristic of each alternative divertor concept with a constraint on the plasma-wall distance and on the plasma elongation. The feasibility of the configurations is evaluated in terms of maximum vertical force and current density on the PF coils at the start of the current flat top (SOF) and at the end of the flat top (EOF). A 2D and 3D vertical stability analysis is also performed to evaluate the effectiveness of the configurations.

[1] R. Albanese, et al., Fusion Engineering and Design (2017),
https://doi.org/10.1016/j.fusengdes.2016.12.025

[2] R. Ambrosino, et al., Fusion Engineering and Design (2017),
https://doi.org/10.1016/j.fusengdes.2017.01.055

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P3.103 Manufacturing and characterization of stainless steel / CuCrZr and stainless steel / Cu transition ring by explosive welding method

Actively cooled plasma facing components (PFC) are often made of CuCrZr, whereas the cooling pipes are made of stainless steel. Both materials are not easily joined, and a common solution is electron beam welding, using a ring made of Inconel or Ni as intermediate.

This paper reveals the potential of using explosive welding as an alternative joining technique for multi-material transitions of CuCrZr/stainless-steel. Two plates of 16mm thickness were assembled by explosive method. Then rings (13.7mm outer diameter, 10.4mm inner diameter) were machined out from this assembly. These rings were finally welded onto the PFC (CuCrZr) by electron beam and on the cooling pipe (stainless steel) by TIG welding.

Thermo-mechanical simulations were done to determine the stress level around the joint, for a typical WEST PFC and different scenarios (baking and normal operations in WEST). A design load case was then determined.
Afterwards, mechanical tests were carried out on relevant sample geometry for a significant stress level. Tensile tests were done at both room temperature and 200℃ to determine the ultimate tensile strength and the maximal elongation. Rotative bending fatigue tests were also done to ensure that the CuCrZr / steel joint had a sufficient lifetime regarding the expected operation. The quality of the sealing was also tested with He leak testing procedure at high temperature.
Finally, a second type of ring made of Cu / steel was also considered. It is shown that due to the explosive process, Cu is largely hardened, and thus also suitable as structural material. Moreover, welding of Cu on CuCrZr is more reliable than CuCrZr on CuCrZr, making this solution even more appropriate.

The paper concludes on the adequacy of using explosive welding for multi-material transitions of CuCrZr or Cu and stainless-steel, for linking a CuCrZr PFC to a stainless steel cooling pipe.

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P3.010 Design and manufacturing of thermal shield for JT-60SA

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In order to reduce thermal radiation to cryogenic components, vacuum vessel (VV), ports and cryostat are covered with thermal shield in JT-60SA. The design requirements are follows;
- Surface temperature shall be < 100K during operation and < 140K during VV baking.
- It should have integrity against electro-magnetic force during plasma disruptions and seismic load (0.6G in horizontal and 0.4G in vertical).
- It should have enough clearance for displacement of each components.

Thermal shield is designed to satisfy above requirements as follows. It consists of VV thermal shield (VVTS), port thermal shield (PTS) and cryostat thermal shield (CTS). They are fabricated each toroidal 20 degree sectors in factories and assembled on-site. They have double wall structure made of SS316L. Cooling pipes between walls are cooled by 80K He gas at 1.8 MPa. Each sectors has two cooling channels for redundancy. Outer surface of CTS is covered with multi-layer insulations. VVTS and CTS have electric insulations at each 20 degree in toroidal direction to reduce electromagnetic force. Vertical load is hung from toroidal field coil cases by 18 rods. Horizontal load is supported from cryostat body with 27 stays. Special shapes are adopted at 8 of 18 sectors to keep required clearance to tangential horizontal ports and oblique ports for NBIs.
The VVTS and PTS except for 8 special shape horizontal PTS have been fabricated. Fabrication tolerances were within +/-5mm for inboard VVTS and PTS, +/-10mm for outboard VVTS. Pressurized test at 2.5MPa and He leak tests less than 10-8 Pam3/s were passed twice, after welding between cooling pipes and inner plate and after closing outer plate in factories. 340 degree of VVTS and lower PTS have been assembled on-site. Each 20 degree sector were connected customized mechanical couplers and radiation covers to keep required clearance around them.

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P3.012 Design of Integrated Bivalent Auxiliary System (SABI) supporting the MITICA cooling plant and cryogenic plant

The first ever built full-scale prototype of the ITER heating neutral beam injector is the MITICA experiment at PRIMA Neutral Beam Test Facility, under realization in Padua, Italy.
This experiment consists of several in-vessel components: most of them are actively cooled by a large cooling plant, still under construction. Coolant is deionized water produced by a Chemical Control System (CCS), part of the cooling plant.
Furthermore, two large flat-geometry cryopumps will be installed in the MITICA Beam Line Vessel in order to reach the required vacuum performances; the cryopumps will be supplied with ScHe and GHe produced by a cryogenic plant, now in manufacturing design phase.
During the installation of cooling plant and the design phase of cryogenic plant, two different issues came into evidence: sewage water (produced by the CCS during operations) could not be delivered directly into the public network; warm compressor of cryogenic plant, originally air-cooled, changed to a water-cooled system.
For the CCS issue, the solution was found in the realization of a sewage water treatment system (WTU) that requires the supply of hot water to distil the wastewater and to concentrate it.
Opposite, the need to balance the heat loads and the cost and space constraints lead to an innovative system: it is essentially based on a high temperature water-water heat pump in which the heat rejected by the warm compressor is used, by means of this heat pump, to heat up the WTU.
The paper describes in detail this nice example of energy saving.

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P3.013 Status of development and commissioning of the TCV EC-system upgrade

The upgrade of the EC-system of the TCV tokamak has entered in the realization phase as part

of a broader upgrade of TCV[1]. The first of the two MW-class, dual-frequency gyrotrons (84 or 126GHz/2s/1MW) has been delivered by Thales Electron Devices and the full commissioning and characterization is expected to be completed during the first half of 2018. The design of this gyrotron includes new features such as an optimized launcher for the dual-frequency operation, improved cavity and collector cooling, and a steerable last internal mirror. A novel Matching Optics Units adapted for the dual-frequency operation has also been developed [2]. The successful validation of these new features will strongly enhance the gyrotron development in Europe.

In parallel to the gyrotron development, for extending the level of operational flexibility of the TCV EC-system the integration of the dual-frequency gyrotrons adds a significant complexity in the evacuated 63.5mm-diameter HE11 transmission line system connected to the various TCV low-field side and top launchers.

An important part of the present TCV-upgrade consists of inserting a modular closed divertor chamber. This will have an impact on the existing X3 top-launcher which needs be reduced in size and complexity to facilitate manned entry into the vacuum vessel when the divertor reconfiguration is implemented. This will be realized by mounting two, new, independent X3 top launchers each connected to one of the dual frequency gyrotrons.

The main design features and test results of the gyrotron and MOU, together with specific aspects of the transmission line system and new X3-launchers will be described.

This work is supported by the Ecole Polytechnique Fédérale de Lausanne (EPFL).

[1] A. Fasoli et al., this conference
[2] A. Moro et al., this conference

P3 / 1065

P3.018 Design of the 105GHz electron cyclotron resonance heating system on J-TEXT

To meet requirements of heating and physics experiments on J-TEXT, we are developing a 105GHz/500kW/1s electron cyclotron resonance heating (ECRH) system. With the toroidal field of about 2T for normal discharges on J-TEXT, this system will mainly work at the second extraordinary mode. The ECRH system consists of a Gycom gyrotron with a superconducting magnet and related power supplies, a 30m transmission line based on corrugated waveguides, a quasi-optical launcher, a control system, a cooling system, etc. The designed transmission efficiency is about 85% and the injection angle of the electron cyclotron wave can be adjusted by the movable mirror of the launcher. Considering the transmission efficiency and the power loss of the launcher, about 400kW microwave power will be delivered to the plasma of J-TEXT tokamak.

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P3.015 Numerical analyses and tests for optimized and enhanced heat transfer solutions in DEMO

A set of novel design solutions for high performance cooling systems have been developed and tested by Consorzio RFX, achieving, with experimental tests, the challenging heat transfer conditions foreseen for Heating and Current Drive Systems of present and future nuclear fusion devices.
The project, called Multi-design Innovative Cooling Research & Optimization (MICRO), has the triple objective of:

- Verifying the full qualification of the manufacturing process and assess the cooling performance

of the present solutions applied on the accelerating grids designed for ITER Heating Neutral Beam Injectors and MITICA experiment;

- Identifying alternative designs realizable with presently qualified manufacturing processes for improving heat transfer mechanisms with limited pressure drops;
- Developing further optimized solutions by means of the investigation of the interrelated effects of geometric parameters and thermofluid dynamic properties of the coolant fluid.

The main aim of present work is on one hand to extend the fatigue life-cycle of high thermal stress components and to investigate the possibility to employ alternative dielectric fluids with respect to the present project solution (ultrapure water). Such category of fluids is characterized by poorer thermofluid-dynamic properties resulting in a less performing cooling capability. Such drawback is meant to be compensated by the enhancement of the heat transfer mechanisms guaranteed by the novel design solutions.
If the thermo-structural requirements set for such kind of components would be nevertheless satisfied, these dielectric fluids could represent a significantly advantageous option with respect to the existing technologies.

This paper presents in the first part the results obtained from a series of thermofluid-dynamic tests carried out on sub-scaled models of the electrostatic grids, while in the second the outcomes of numerical analyses implementing perfluorinated compounds, instead of water, and the preparation of next experimental tests.

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P3.016 A high power helicon antenna for the DIII-D tokamak and its electromagnetic aspects

A comb-line antenna to demonstrate efficient off-axis non-inductive current drive from the absorption of toroidally directed very high harmonic fast waves is being designed and built for DIII-D [1]. The antenna consists of a toroidal array of 30 modules, each 5 cm wide by 21 cm tall, so that the array spans 1.5 m on outer wall just above the tokamak midplane.
This antenna will be fed with 1 MW of RF power at 476 MHz [2] by a coaxial feedthrough that transitions to a stripline feed inside the vacuum vessel which divides the input RF power evenly and at the proper phases into the four inputs of the end modules of the antenna. COMSOL was used to optimize the RF performance of the stripline and the RF coupling into and between the modules by varying their geometry and spacing. Multipactor phenomena is being analyzed using the SPARK3D code, which predicts it occurring in the stripline and the modules, motivating changes to the surface properties of the materials used and the RF power levels and toroidal magnetic field at which the antenna can be operated.
In addition, the forces are computed due to the currents induced in the striplines, modules, and their support systems by plasma disruptions. The current magnitudes and directions are plotted so it is possible to identify the locations where the largest current is perpendicular to the toroidal magnetic field. By modifying the antenna support geometry, it is possible to optimize the currents to minimize the resultant forces and torques on the whole antenna system.

[1] R. Prater, et al., Nucl. Fusion 54, 083024 (2014).
[2] A. Nagy et al., Fusion Science and Technology 72, 623-627(2017)
* Work supported in part by the US Department of Energy under DE-FC02-04ER54698

P3 / 1064

P3.017 Conceptual definition of an ICRF system for the Italian DTT

An Italian Divertor Tokamak Test (DTT) facility has been proposed to tackle a major mission of the European roadmap to fusion electricity, i.e., the problem of power exhaust. DTT is conceived to accomplish very different magnetic configurations and to reproduce edge conditions as close as possible to DEMO, allowing a reactor-relevant exploration of alternative power exhaust solutions in an integrated plasma scenario. To attain a power-over-radius ratio of around 15 MW/m crossing the separatrix, a heating power of around 40-45 MW is required and will be provided by a suitable mix of electron cyclotron resonance heating, neutral beam injection and waves in the ion cyclotron range of frequency (ICRF).
The ICRF system is called to couple 5-10 MW of power to plasma in the frequency range 60-90 MHz for pulse lengths of 100 s every 30 minutes, allowing central heating of either 3He or H minority species in the reference D plasma. The system will be also an essential tool for fast particle generation, allowing the emulation of alpha's behaviour and the study of enhanced reaction rates via optimization of particle distribution functions.
This paper reports the status of its conceptual design, providing the rationale behind main design choices as for example the external matching scheme or the power density at the launcher front face. Different antenna concepts are reviewed and RF designs based on two, three and four straps in toroidal direction are compared on the basis of simulation results and performances documented in literature. Strengths and weaknesses of each antenna candidate are discussed to identify the most promising concept for DTT. Innovative solutions, e.g., based on the use of high-impedance surfaces, are considered too.

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P3.020 A new tungsten wire calorimeter for the negative ion source testbed BATMAN-Upgrade

Within the framework of the ion source development for the ITER and DEMO Neutral Beam Injection (NBI) systems, IPP Garching has recently upgraded the radiofrequency-driven negative ion source testbed BATMAN. One of the requirements for the ITER NBI system is to produce a beam power density homogeneity above 90% over its large extraction area of about 0.2 m2. This requirement is going to be addressed, on a smaller scale, in BATMAN-Upgrade with ITER like beam optics. The testbed is equipped with several beam diagnostic tools to measure beam power, uniformity and divergence. A new tungsten wire calorimeter (TWC) has recently been developed to characterize the beam quantitatively and with an improved spatial and temporal resolution. The TWC consists of an array of thin tungsten wires with a diameter of 0.3 mm, placed in the beam path. The wires are heated up by the powerful beam up to 3000 ℃ and emit visible light, which is observed by an optical camera. While the existing TWC provides a qualitative impression of beam power characteristics and are placed at a distance of almost 2 m from the beam extraction system, the new TWC is positioned only 20 cm downstream of the extraction system and should allow observing the single beamlets for the first time in this testbed. In order to determine the correlation between the pixel intensity measured by the camera and the power density impinging on the wires, one wire is equipped with an ohmic heating system which will allow heating up the wire with a known power, thus enabling calibration of the diagnostic tool.
In this paper the design, including FEM simulations of the wire thermal behavior, manufacturing and installation of the new TWC inside the BATMAN Upgrade testbed is described. First results are also presented.

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P3.021 Modeling neutral beam transport in fusion experiments: Studying the effects of reionisation and deflection

Neutral Beam Injection is the main heating system on a variety of fusion devices, and will be the main heating system of ITER. Especially during high-power operation in long pulse devices, it is important that the losses in the beamline are well quantified. There are several types of beamline losses, such as geometrical losses, due to scraping of the beam on apertures, and reionisation losses, due to interaction of the beam with background gas. Geometrical losses are straightforward to calculate, and lead to well-predictable heat loads. In contrast, quantifying reionisation losses requires knowledge of the local gas density, and the resulting heat deposition depends on the magnetic field. Because of the deflection of the reionized particles, heat deposition can occur in shadowed locations. A computational toolbox is developed to calculate these losses. The results are benchmarked on ASDEX Upgrade. The codes will be used to aid NBI design in future devices such as DEMO.
Damage to components in shadowed locations has occurred in the NBI beamline of ASDEX Upgrade. To model the heat deposition due to reionisation, the vacuum magnetic field in ASDEX Upgrade is calculated for various discharges to identify critical magnetic field conditions in terms of duct damage. For selected shots, the magnetic field including shielding materials and saturation effects is calculated with ANSYS. The neutral density in the NBI beamline of ASDEX Upgrade is calculated assuming molecular flow, taking into account the full 3D geometry and the properties of the pumps. Results from the neutral gas density calculation are benchmarked with experimental pressure measurements. To calculate heat loads, neutral beam particles are tracked through the magnetic field while interacting with the neutral gas via a Monte-Carlo scheme. The damage pattern observed in the experiment is explained by reionisation and subsequent deflection under specific magnetic field conditions.

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P3.031 DONES Central Instrumentation and Control Systems: general overview

Purpose of this work is to describe the DONES Central Instrumentation and Control System (CICS). A functional definition of the main systems will be given, together with a general overview of the current status of the CICS and the differences with respect to the corresponding system developed during the IFMIF-EVEDA phase. The overall architecture of the Control System, the definition, functions and requirements of the Control System, the identification of Subsystems and equipment, the control system design guidelines, the interfaces and definitions of CICS and other Systems, and the Control Room requirements, will be described. Moreover, the DONES Local Instrumentation and Control System (LICS) will be defined and the relationship between CICS and LICS described.

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P3.026 Achievements of the ELISE test facility in view of the ITER NBI

Neutral Beam Injection (NBI) for ITER shall deliver in total 33 MW heating power to the plasma with two injectors at a beam energy of 1 MV. Taking neutralisation efficiency and all losses along the beam path into account a negative ion current density of 329 A/m2 (H- for 1000s) and 286 A/m2 (D- for 3600s) has to be extracted from each ion source (size 1 × 2 m2) with a beam non-uniformity below 10%. For the development of the radio frequency driven (RF) ion source and the extraction system the ELISE test facility has been set up at IPP as a first step with a source size of 1 × 1 m2.

Since its start in 2013 continuous progress was achieved hardening the system technically for long pulse operation at high power (300 kW) as well as approaching ITER´s physics requirements. 90% of the required current density could be demonstrated for short pulses (10 s) and almost 70% for long pulses, both in hydrogen and deuterium. The main limitation is given by the amount and temporal instability of the co-extracted electron current, in particular in deuterium, which has to remain below the negative ion current to prevent thermal overloading of the grid system.
Shaping the magnetic field topology and controlling the electric fields inside the source together with an improved caesium conditioning procedure and a plasma grid temperature above 150℃ have been identified as possible tools to increase the source performance. Experiments in 2018 concentrate on optimising these parameters for long pulses, mainly in hydrogen. Furthermore emphasis is also laid on how to improve beam divergence and homogeneity.
The paper presents the recent results of ELISE and discusses the latest results and possible modifications which might be required for the ITER source.

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P3.030 Optimization of TE11/TE04 Mode Converters for the Cold Test of a 250 GHz CARM Source

Electron Cyclotron Resonance Frequency (ECRF) systems in future fusion devices, like the DEMOnstration reactor, foresee an operational frequency in the range 230-280 GHz to match the plasma characteristics. The Cyclotron Auto Resonance Masers (CARM) characterized by a high value of a frequency Doppler up-shift, could represent an alternative to gyrotrons and the design of a 250 GHz, 0.5 MW CARM device has been undertaken by ENEA. A mode conversion from the fundamental TE10 mode in WR-3 rectangular waveguide to the TE53 operational mode in an oversized circular waveguide is required to perform the cold test of the two Bragg reflectors delimiting the CARM resonant cavity. An improved structure for the TE11 to TE04 conversion, using only two transitions with profile and radius optimized in order to obtain high efficiencies and sufficient bandwidth, has been designed and presented in this paper. The first transition is a TE11/TE01 serpentine mode converter in circular waveguide with average radius of 1.48 mm and an appropriate number of geometrical periods. The second one is a TE01/TE04 rippled wall mode converter with an average radius of 3 mm. It consists of a single conversion section with sixteen sinusoidal periods instead of three conversion sections (TE01/TE02, TE02/TE03, TE03/TE04) in cascade, thus reducing the complexity and the length of the full transition chain. The last conversion (TE04 to TE53), a rippled-wall mode converter, made by a helically corrugated waveguide, has been described in a previous paper. The analysis with HFSS and CST- MWS (S-parameters, efficiency and electric field) of each single converter will be reported; for the TE01/TE04 transition, results obtained by in-house codes will be shown too. A first validation of the simulation tools at 250 GHz will be carried out through the construction of a short Bragg reflector during the current year.

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P3.034 Disruption simulation for design of JT-60SA components.

Disruption simulations with DINA code are performed for JT-60SA design. The simulation results have been applied to design of many components, not only for the vacuum vessel and in-vessel components, but also for peripheral components. For instance, for design of in-vessel coils, the stabilizing plate and magnetic sensors, EM force induced by hallo current and eddy current at disruption were calculated. For design of poloidal field (PF) coils, the power supply of PF coils and refrigerator system for super conducting coils, eddy current of PF coils and AC loss of superconducting coils were evaluated. DINA code calculates free boundary plasma equilibria, taking into account

eddy currents in the vacuum vessel and a model of the power supplies. We performed three type of disruption simulation, up and downward VDE disruption and major disruption (MD). VDE is caused by vertical instability due to loss of control. As for MD, Plasma stays at center when disruption starts. Poloidal halo current is larger with longer current quench time and this result is same as that done for ITER. We will present details of JT-60SA DINA simulation and application of them to JT-60SA design.

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P3.036 NI p2p streaming powered multiple fpga real-time data processing system on J-TEXT tokamak diagnostics

On tokamaks, there are many diagnostics, which need real-time data acquisition and processing to provide useful information for plasma control. Some of the diagnostics required fast processing of multiple very high sampling rate signals. It is often difficult to achieve even with modern multi-core CPUs. This is due to moving large amount of data from the digitizers to the system ram would hurt the deterministic performance significantly. FPGA data processing is a better choice when speed and deterministic performance is required. When the task is too heavy for a single FPGA to implement, transfer large data between FPGAs is also required. On the polarimeter-interferometer diagnostics on J-TEXT, in order to calculate the density profile of the plasma in real-time, multiple chords of line-integrated density must be calculated first. The data acquisition system for the polarimeter-interferometer is equipped with 5 FlexRIO FPGAs, only one is acquiring the reference signal. Density calculation for every probing chord needs the reference signal. So this signal needs to be transfer to other FPGAs with strict real-time requirement. Mismatching a single sample the result would be completely incorrect. NI P2P streaming allow the FPGA to transfer data without burdening the process. In the paper, the determinism characteristic and other performance of the P2P streaming is tested, and a mechanism to align local digitized samples with P2P streamed samples is presented. These results can be applied to other real-time data processing system that requires multiple FPGAs.

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P3.039 Designing CODAC System for Tokamaks Using Web Technology

ITER CODAC is the most sophisticated tokamak control and data acquisition system. The core of ITER CODAC is built around the EPICS toolkit. EPICS is very mature in accelerator community. However, there are still works trying to improve existing control system software like tango and EPICS 7 mainly driven by the needs of more flexible system and development of computer technology. This paper present a new way of building a CODAC system for tokamaks using web technology instead of EPICS toolkit. The control system components are abstracted into resources. The accessing of the resources is done via standard HTTP RESTful web API. HMI is based on HTML and JavaScript in browsers. WebSocket is used to improve the efficient of event distribution. The main feature of this design is all the interfaces in the system are based on open and industrial web standards, which are interoperable among almost all kinds of server and client technology. The paper also presented a software toolkit to build this CODAC system. With it, a data acquisition system for ECEI diagnostics was built and presented.

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P3.040 Requirements management support for the ITER PCS in view of first plasma operations

The Plasma Control System (PCS) is one of the main ITER systems. It is in charge of running the plasma discharge, by receiving data from the real-time diagnostics, and by computing the commands to be processed by various plant systems to act on the plasma (e.g., the power supplies of the poloidal field coil circuits and the additional heating systems).
To this aim, the PCS will implement several different functions, which will be not limited to control algorithms, but they will also include support functions that process the inputs acquired from different diagnostics, and exception handling. There are many different types of requirements for the ITER PCS (functional, architectural, operational, etc.), which come from a heterogeneous set of sources. Moreover, the design of some plant systems the PCS has to interface with is not yet finalized, and the way ITER will be operated is also under development.
In such a complex context, it is essential to rely on a system-engineering oriented approach for requirements management, to both track the development of the PCS and to assess the compliance of the product with the requirements set expressed at each operational phase (first plasma operation, D-T operations, etc.). The current ITER choice is to use of the so-called PCS Database (PCS DB) that acts as the central collaboration tool for the system design and is deployed using Enterprise Architect. The PCS DB not only traces system requirements to higher level ones, but also maps PCS functions to the architectural components and to simulation models, in order to document the set of the PCS functionalities and prototypes.
In this paper, the overall methodology for the PCS requirements management process, and the current status of the PCS DB will be described, focusing on the elements relevant for first plasma operations.

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P3.043 Development of the KSTAR Alarm Management System

The control systems for the Korea Superconducting Tokamak Advanced Research (KSTAR) have been implemented based on the Experimental Physics and Industrial Control System (EPICS) which is a framework for control systems widely used on accelerators and fusion devices including the ITER, an international fusion experiment.

In terms of operation and maintenance of KSTAR control systems, there was a need to have an alarm management philosophy for the KSTAR and to implement an alarm system compliant with the philosophy for comprehensive and systematic monitoring of device, efficient and timely operator's action against abnormal situation.

Therefore the KSTAR control team has developed a guideline for alarm management for the KSTAR, and implemented an alarm system following the guideline by using the EPICS based CS-Studio alarm functions.
The KSTAR alarm management system has been deployed using the Docker which is a container based virtualization technology for ease of deployment and management and better performance in comparison with virtual machine.

This paper describes a summary of the KSTAR alarm management guideline and implementation of the KSTAR alarm management system.

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P3.044 Sensitivity of fast ion losses to magnetic perturbations in

the European DEMO

The limits for the heat loads on the DEMO first wall are significantly stricter compared to those of ITER due to cooling and breeding blanket requirements. In addition to the thermal particle and radiation loads, fast particles in the form of fusion alphas and NBI ions with high energies can escape the confinement due to various magnetic perturbations and produce a significant heat load on the first wall.

Previously, the losses of fusion alpha and NBI ions have been found to be manageable, with wall loads remaining below 10% of the limit of 1 MW/m2 envisaged for the DEMO first wall. This is primarily due to the plasma profiles that allow the 800 keV NBI ions to penetrate deep into the plasma before ionizing, and the generous gap of up to 20 cm between the separatrix and the first wall panels. Additionally, only losses due to toroidal field ripple were accounted for, while significant perturbations would be introduced by the possible addition of ELM control coils.

In this contribution, we present ASCOT simulation results for fast ion confinement and losses under the effects of additional perturbations, including increased toroidal field ripple due to reduced ferritic inserts and reduced number of toroidal field coils, and the addition of resonant magnetic perturbations (RMP) due to ELM control coils. Additionally, the effect of NBI ionization in the scrape-off layer is discussed, which can introduce localized loads due to high-energy prompt losses.

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P3.045 Simulation of magnetic control of the plasma shape on the DEMO tokamak

This paper describes the preliminary design of a position, current and shape control for DEMO tokamak. This preliminary design relies on the availability of magnetic sensor measurements for the vertical position and for the plasma-wall gaps. The controller is designed basing on the CREATE-L model of the DEMO 2017 Single-Null (SN) configuration, and then is tested using the nonlinear evolution CREATE-NL model. The controller is designed along the guidelines presented in [1], and it consists of various nested loops: i) a Vertical Stabilization fast controller which stabilizies the plasma keeping the current in the stabilization circuit as low as possible; ii) a shape controller which calculates the Poloidal Field (PF) feedback currents that are needed to track a given reference shape or to keep the shape as constant as possible in the presence of unexpected events; iii) a PF current controller which evaluates the PF voltages needed to track the feedback currents calculated by the shape controller; iv) the plasma current controller which keeps the plasma current close to its reference.
The controller performance is assessed in the presence of a given set of events: i) the Vertical Displacement Event (VDE) of the plasma column, which consist of an instantaneous vertical displacement along the eigenvector associated to the unstable mode; ii) a so-called loss of power which occurs during a plasma H-L back-transition; iii) NTM; iv) ELMs. In the closed-loop simulations, simplified yet realistic models of the actuators, including saturation values for the PF voltages, and of the magnetic diagnostics are included.
The paper will include the main simulation results of the behaviour of the DEMO SN configuration in the most challenging events as listed before, highlighting the most critical issues.

Reference
[1] M. Ariola and A. Pironti, “Magnetic control of tokamak plasmas,” 2nd ed., Springer, 2016.

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P3.046 Towards an automatic filament detector with a Faster R-CNN on MAST-U

Plasma behavior in the SOL of tokamaks is driven by turbulence in the edge region where density and temperature gradients are large. This generates intermittent structures of increased density and temperature known as filaments, which extend along the magnetic field lines. The protection of plasma facing components in the next step devices is a primary concern. In this context, the identification of the filaments, for the understanding of their generation and propagation schemes, is a main issue.
The goal of this work is to develop an automatic detector for filaments arising in the MAST-U plasma. The identification of the filaments must be done starting from the 2D images acquired with a fast camera. To this end, the potentiality of the Faster R-CNN, which are suitable for image classification and object recognition, was investigated in [1].
In the present paper, a database of several thousand of images generated by a synthetic diagnostic, which reproduces the statistical properties of experimental filaments in terms of position and intensity, has been used. The synthetic images have been pre-processed by approximately mapping them onto the toroidal midplane of the machine. Preliminary results show a high detection rate (more than 0.85) for filaments with high pixel intensity and large area. Indeed, a more suitable definition of the training target box is needed to improve the capability of the detector in identifying filaments with low pixels intensity, small area or located on the edges of the frames. Therefore, to enhance the detector performance, a new definition of the target box has been deeply investigated, by considering the intensity and the size of the filament.
[1] B. CANNAS, et al. (2017). Convolutional Neural Networks for the identification of Filaments from Fast Visual Imaging Cameras in Tokamak Reactors. In Neural Advances in Processing Nonlinear Dynamic Signals, Springer.

P3 / 1095

P3.048 Study of the methods to enhance infrared emissivity of Molybdenum surface

Single crystal Molybdenum is one of the most promising materials for the First Mirror (FM) for ITER optical diagnostics due to high resistance to erosion under the neutral atom bombardment. Other advantages are: low CTE, high thermal conductivity, good mechanical properties at elevated temperatures. The FMs are normally located in the front-end of ITER port plugs, being subject to the volumetric heat loads up to ~1W/cm3 and higher. Active cooling of Mo mirrors by the water or gas flow has limited applicability due to remote handling, integration and risk requirements. So-called athermal design with radiative cooling is developed for the H-alpha FM unit with all-Mo structure comprising the housing, two mirrors and cleaning electrodes. Main idea is to minimize the mirror displacements and to keep the alignment at the temperatures up to ~350℃. That requires the balance between thermal contact and radiative heat sinks. Most uniform profile is obtained by weak thermal contact between FMU and support structure, but it leads to the highest FM temperature, since the normal as-milled Mo surface has very low effective emissivity <10% at 200..300℃. The well-known tools for the surface blackening: enhanced roughness, oxidation, V-grooving, coatings, must be analyzed for the outgassing rate, thermal, radiation and long-term stability in Hydrogen environment.
In this study, a number of the techniques to enhance Mo surface effective emissivity have been tested: V-grooving, detonation spray coating and surface electro-erosion. Spectral emissivity of the test samples have been measured by Bruker Vertex-70 infrared Fourier spectrometer and the effective values are derived for the subsequent thermal analyses. Alumina coatings were found to be the most effective tools with emissivty ~80%. However, the V-grooving technique also gives the acceptable result ~25..30% and it does not require the qualification for applicability to ITER. The trade-off between different techniques is discussed in detail.

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P3.049 New data acquisition control software developments for the simultaneous control of ten intelligent overview video cameras at W7-X

In the past two campaigns of Wendelstein 7-X stellarator the overview video diagnostics played an important role in the daily experiments. The current software implementations went through numerous improvements and changes according to the continuously changing requirements. However, while the control software could handle all the needs, the changes reached a point where the redesign and re-implementation became necessary to ensure high reliability. In the next campaign of W7-X real-time data streaming will be implemented to W7-X's Archive, which can be practically a limitless real-time data backup. Other improvements involve high-level interaction with the Trigger Time Event (TTE) network, intense Graphical User Interface (GUI) improvements such as the grouping of cameras, visual display of measurement settings (eg. exposure and readout timing). This new software package is arranged into two standalone pieces, similarly as in the previous version. One of the main upgrades is that the software will be able to send so-called event descriptions over the network during the measurement to a central event display, where events from several diagnostics will be visualized. This feature can be very important in physics measurements because users can visually see what is happening inside the machine in real-time.
This paper will present the detailed experiences, new requirements and detailed design, implementation and testing of this new software package.

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P3.050 Structural analysis of the ITER ex-vessel PPR components located in the Gallery

The Plasma Position Reflectometry (PPR) diagnostic system (PBS 55F3) is planned to provide information related to the edge electron density profile and plasma position at four defined locations distributed both poloidally and toroidally in the ITER vacuum vessel.

The sections of the ex-vessel transmission lines (TL) in the Gallery, between the two secondary confinement barriers, are considered "captive components". They are installed as close as possible to the ceiling, so that they are trapped by other systems installed below. Consequently, these sections must be installed before any other system running underneath and will not be accessible for replacement or repair once all these systems are installed.

The PPR system consists of five fully independent reflectometers grouped in three different paths, depending on the port through which are routed (Equatorial Port 10, Upper Port 01 and Upper Port 14), with different lengths and waveguide distributions. The waveguides in the Gallery are supported by several independent structural frames which are attached to the building. All the frames in the same route are quite similar among them, but they differ from those in the other routes.

The structural analysis strategy, based on this similarity, is to build a global model for each PPR route in the gallery in order to analyze global deformations, reactions on supports and identify the critical parts of the whole PPR system.

Based on the results obtained in the global model, detailed model has been built for critical components to carry out the thermo-mechanical analyses of every load combination and asses the structural integrity according to RCC-MR code.

P3 / 1100

P3.053 Design of inductive sensors and data acquisition system for magnetic diagnostic of the T-15MD tokamak

Magnetic diagnostics plays an important role in tokamak operation. Magnetic data are used for real-time control of plasma current, shape and position and for post-discharge analysis of magnetohydrodynamic plasma instabilities and equilibrium reconstruction. The magnetic diagnostic of the T-15MD will consist of more than 500 inductive sensors of various types: poloidal flux loops, saddle loops, Rogowski loops, diamagnetic loops, magnetic probes. Location and number of sensors is determined by the measured physical value. Most sensors, including 6 poloidal sets of 48 high-frequency (up to 300 kHz) magnetic probes for investigation of three-dimensional structure of Alfvén eigenmodes with high toroidal and poloidal mode numbers (m,n≤16), will be located inside the vacuum vessel.
The data acquisition system must simultaneously provide both some sensors data real-time transfer to the control loop and all sensors data storage for subsequent analysis. The requirements for the data acquisition depend on the data destination. Since the duration of the plasma magnetic control cycle (determined by the characteristics of the tokamak and the power supply system) for the T-15MD is about 1 ms, it is sufficient to have a sampling rate up to 10 kHz for the data transferred to the control loop, but these data must be transferred and processed in real-time mode. At the same time, for the study of fast processes (disruptions, tearing modes, Alfvén eigenmodes), sensor signals should be acquired with a sampling rate from 100 kHz to 1 MHz (depending on the type of sensor) and transferred to the database at an acceptable time (about 3 minutes) after discharge.
The article describes the construction and location of inductive sensors, as well as the architecture of the data acquisition system.

P3 / 1101

P3.054 Real-time waveform classification in TJ-II Scattering Thomson diagnostic using ensemble methods

Over the years, humanity has needed energy constantly due increase both population and the technology. That's why conventional methods of energy production are not enough to cover new demands especially in environmental area because of pollution generated.
Energy generated by nuclear fusion solve both problems so achieve full control over it internal process, this implies an analysis over huge databases with different kind of signals. Of course it is impossible to do it manually and is essential automate process.
One of the multiple ways to achieve the above goal is generating machine learning models. It just needs input data and an answer for each input. We can find several algorithms to carry out classification in the literature in order to build such models. Most popular algorithm are support vector machines and neural networks, both have shown high performance in previous applications in fusion, but with an inconvenient: it's impossible to know the reason about why the algorithm decide between one class or another.
Ensemble methods provide good balance among success rate and internal information about model. Particularly, the Adaboost algorithm will allow to obtain an explicit set of rules that explains the output (the class) for each input data.
In this paper an innovative method is proposed where algorithms commonly used in image classification were used to classify waveform of nine different classes of the TJ-II stellerator fusion device located at Madrid (Spain). Average success rate of the model was 97.57%, improving the results obtained in previous works were neural networks and support vector machine techniques were used. In order to take advantage of all features of Adaboost algorithm, a sensitivity analysis were also performed to select a reduced set of features in order to classify the waveform before it actually finishes which could be very interesting for real-time applications.

P3 / 1103

P3.056 Measurement of edge plasma parameters at W7-X using Alkali Beam Emission Spectroscopy

A 60 keV neutral Alkali beam system was designed, built and installed for beam emission spectroscopy measurement of edge plasma on W7-X.
The injector consists of three parts: a recently developed thermionic (lithium or sodium) ion source (j≥2.5mA/cm2), a high focusing efficiency ion optic (~50% of the extracted current can be found in the plasma) and a newly developed recirculating neutralizer; it gives the possibility to measure continuously even at long (<100s) discharges.
The observation system consists of two parts: a 40 channel avalanche photo diode (APD) camera unit and a CMOS camera which run in parallel: 95 % of the collected light goes to the APD unit which is digitized with 2MHz sampling rate while the CCD camera is operated in the 100 Hz range.
As the decay length of the Sodium first excited state is much shorter than Lithium it was selected as beam species. This way the steep density gradient at the bean shaped cross section of the Wendelstein 7-X plasma can be resolved.
The diagnostic beam can be removed from the plasma using deflection plates driven by a fast high voltage switch. This beam chopper can be operated up to 250 kHz giving a possibility to monitor the background on the time scale of the turbulence.
Unexpectedly a thermal sodium beam was also observed in the far Scrape Off Layer in the W7X plasma during the first measurements. It was found to originate from the recirculating neutralizer. Its light signal is at least one magnitude higher than the local accelerated beam signal, although its penetration is limited to the far SOL. To resolve this issue potassium filled neutralization experiments are being performed.
In this paper the main improvements of the sodium beam system and the first measurement results are described.

P3 / 1104

P3.057 Geometrical effect in the measurement of the CXRS on EAST

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Charge eXchange Recombination Spectroscopic (CXRS) diagnostic system was successfully applied on EAST campaign. The CXRS located at D port was designed to focus on the tangential neutral injection beam from Port A on EAST. However, the tangential beam and the perpendicular beam are always injected into the plasma at the same time for the better heating and current driving. Therefore, spectrum collected by the CXRS contains information from both beam, which deteriorates the spatial resolution of the CXRS. To solve the problems, the simulation based on the simulation of spectra (SOS) was carried out. An upgrade on SOS has been performed to add the effect of the width of the injected beam. Based on the improved SOS, the simulation work has been done with input parameters (beam, geometry, and spectrometer parameters based on the actual EAST data; plasma properties were shaped as: ne, Te, Ti, Vt∝(1-(r/a)^2)^α). The results showed that if the active spectra were fitted by single Gaussian, the deviation brought by geometrical effect relies on the gradient of the plasma parameters, for example, up to 30% error at the edge and ~20% at the location of internal transport barriers (ITBs) on ion temperature and rotation measurement, in where the gradient of the parameters are large, while the error in the core without ITBs is less than 1%. Based on the simulation results, an iteration code was developed to correct the error, and make a good consistency with the input ion temperature and rotation profiles.

P3 / 1105

P3.058 Upgrade of the beam emission spectrometer on EAST

The beam emission spectrometer that shares the same collection optics with the existing core charge exchange recombination spectroscopy (cCXRS) on EAST has recently been upgraded. The enhanced

system allows the simultaneous measurements of red- and blue-shifted parts of the Doppler spectrum as well as the active charge exchange line (Dα n = 3-2 656.1nm) from the main ions. One curved strip on the glass made by the method of optical mask is used to attenuate the cold Dα emission line. The operating spectral bands of the system is 656.1±6nm, considering the Doppler shift induced by the maximum angle between the line of sight and neutral beam injection (NBI) as well as the maximum injection energy of 80 keV for deuterium beam. The enhanced system aims at the simultaneous measurements of the neutral beam emission from the Co-current and Counter-current NBI as well as the active charge exchange line (Dα n = 3-2 656.1nm) from the main ions. In this paper, an overview of the upgraded spectrometer will be shown and the first experimental results are presented.

P3 / 1107

P3.060 Linux device driver for Radial Neutron Camera in view of ITER long pulses with variable data throughput

The ITER Radial Neutron Camera (RNC) Data Acquisition (DAQ) prototype is based on the PCIe protocol as the interface to be used between the I/O unit and the host PC, enabling for the scalability of the final RNC DAQ system and allowing a sustainable 2 MHz peak event to cope with the long plasma discharges, up to half an hour.
The high performance computer receives the acquired data through the Direct Memory Access (DMA) channels at maximum data throughput of ~1 GB/s per digitizer and for each data block written in the DMA, the hardware generates an interrupt.
To provide the interface between the hardware and the host applications, a Linux device driver was developed, which receives the data from the hardware and stores it in an internal memory. In the top level of the system, there are high performance applications that read the data and processes it accordingly in real-time, such as the neutron emissivity profile calculation or the pulse data compression.
The preliminary tests show that the Linux kernel miss some hardware interrupts when the time between interrupts are in the microsecond scale, which implies data loss if a traditional read technique is implemented. The usage of the pooling mechanisms is not suitable as the variable event rate over the same discharge will not allow to define an optimum fixed time to retrieve data.
This contribution presents the architecture of a different device driver approach that improves the performance and reliability, implementing an internal data transmission correction algorithm. This enables the device driver to automatically check and recover missing data blocks, recovering the data losses in a transparent way for the host applications. The performance results for tests with different event rates and different acquisition time up to half an hour are also presented.

P3 / 1110

P3.063 Design manufacturing and testing of a Long Focus Spectro-Telescope mockup for H-alpha diagnostic

H-alpha and Visible Spectroscopy is one of the ITER first-plasma optical diagnostics providing full poloidal coverage of plasma scrape-off layer (SOL)by two poloidal-view channels in EP11, one tangential-view channel in EP12, and one divertor-view channel in UP02. The diagnostics is composed of several optical sub-units, which transfer the SOL image to the narrow-band filtered cameras located in the Port Cell and to the spectrometers in the diagnostic building. One of major subunits of the optical chain is the Long-Focus Spectro Telescope (LF-ST) which main functions are the transfer of the optical beam through the Port Interspace zone as well as the retention of the image quality in case of shifts and tip/tilts of the optical beam due to position variation of the VV or the in-vessel optical elements respectively.
The LF-ST is composed of 2 aspherical mirrors (TM1 and TM2) in an all-aluminum athermal layout

(housing and mirrors). The telecentric design of the LF-ST is able to compensate displacements of the nominal optical beam of up to +/-10mm & tip-tilt variations of up to +/-0.5°. The LF-ST is located in the Interspace Support Structure (ISS) and is equipped with a hexapod for non-operational pre-alignment of the LF-ST.
Based on the H-alpha PDR design a full-scale LF-ST mock-up has been designed, analyzed, manufactured, assembled and aligned. This paper summarizes the design and analysis process and describes the pre-alignment concept and execution. Final optical alignment within the complete optical chain of the H alpha components is presented in another contribution(1) to this meeting.

(1) 'Testing of the optical chain mock up of H-alpha and Visible Spectroscopy for ITER', A. Gorshov et al., submitted to SOFT-2018

P3 / 1114

P3.067 Overview of optical designs of the Port-Plug components for the ITER Equatorial Wide Angle Viewing System (WAVS)

The equatorial visible and infrared Wide Angle Viewing System (WAVS) for ITER is one of the key diagnostics for machine protection, plasma control and physics analysis. To achieve these objectives, the WAVS will monitor the surface temperature of the Plasma Facing Components (PFCs) by infrared (IR) thermography (3-5µm range) and will image the edge plasma emission in the visible range. It will be composed of 15 lines of sight installed in four equatorial ports (no. 3, 9, 12 and 17) in order to survey at least 80% of the overall area of the vacuum vessel.
On top of leading a European consortium including CIEMAT (in Spain) and Bertin Technologies (in France), CEA is currently developing the port-plug components of this diagnostic.
The design of port-plug components has to cope with both challenging performance (wide field of view, diffraction-limited, ...) and severe constraints such as harsh nuclear environment, complex interfaces (the components are embedded within the ITER Diagnostic Shielding Module, DSM), first mirrors facing the plasma, an optical relay transferring the optical beam to the interspace through the vacuum windows while limiting the neutron flux passing through.
This paper provides a comprehensive description of the baseline optical design of the WAVS diagnostic in the Port-Plug and compares it with alternative design solutions. This comparison allows for assessing the most suitable design meeting both the performance requirements and environmental constraints.
For the baseline design, the final performance of the Port-Plug components in operation are assessed thanks to an end-to-end analysis, which includes theoretical aberrations, misalignment, manufacturing errors, thermo-mechanical deformations of the optical surfaces.

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P3.068 The Radial Neutron Camera DAQ Prototype Performance Results

The ITER project aims at building an experimental fusion device, twice the size of the largest current device in operation, JET, to demonstrate the scientific and technical feasibility of fusion power. Presently, ITER is being equipped with a set of diagnostics to provide accurate measurements of plasma behavior and performance, as the neutron diagnostics. In particular, the Radial Neutron Camera (RNC) diagnostic aims at demonstrating the generation of fusion power by measuring the total neutron source strength with high accuracy and reliability, and predicting the feasibility of thermonuclear plasma self-sustained ignition, which depends on the highly energetic ⍰ particles produced from fusion reaction.
The demanding ITER operating conditions will present new challenges to the diagnostics, resulting in high quantities of data production: longer plasma discharges and higher neutron and gamma

fluxes.
The dedicated Data AcQuisition (DAQ) system must be capable of providing sampling rates fast enough to digitize fast-decay time events from the state-of-the-art detectors while processing and sending in real-time the relevant data to the host, without losing events or exhausting the system storage capabilities.
The first accepted challenge was defining and setting up a DAQ prototype dedicated to ITER RNC with the adequate architecture and processing techniques that will not be obsolete in 7 years when a complete system will be built.
The prototype goal is to mitigate known technical risks and detect hidden risks. At the Frascati Neutron Generator facility the real-time pulse processing algorithms, including neutron count, have been evaluated as compliant with the control loop cycle of 10 ms for the neutron emissivity reconstruction. The 1 GB/s of expected data throughput has been achieved using a ×8 PCIe Gen 2.0 link from the digitizing unit to the host computer. The prototype architecture, results and conclusions are presented.

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P3.069 - Study for the microwave interferometer for high densities on COMPASS-U tokamak

The present COMPASS tokamak at the Institute of Plasma Physics in Prague is equipped with the 2-mm interferometer, which gives a possibility to measure line average electron densities up to 1.2x1020 m-3 (the critical density for the interferometer probing waves is 2.43x1020 m-3). A high magnetic field tokamak, COMPASS-U [Panek et al., Fus. Eng.des. 123 (2017) 11-16], will be designed and built there replacing COMPASS. Higher plasma densities in COMPASS-U about 5x1020 m-3 will require a new design of the interferometer. A new microwave interferometer scheme will provide precise measurements in a wide density range and allow using a real-time gas puff density feedback. A solution for real-time line-average electron density measurements based on the solid-state elements is proposed. Among the main features of this system will be use of the two microwave transceivers with close sub-millimeter frequencies and the principle of the "unambiguous" measurement [Varavin M. et al., Telecommunications and Radio Engineering 73 (10) (2014) 935–942]. A study of both vertical and horizontal orientations of the interferometer with respect to plasma shape and mechanical vibration influence for the plasma density measurement is presented. Ray tracing calculations using the FIESTA-8 code are shown for both configurations, demonstrating the propagation of the probing waves through COMPASS-U plasmas. The plasmas (equilibrium, profiles) are going to be simulated using METIS and that we use a range of plasma densities. An assessment of signal corrections corresponding to different vertical and radial plasma positions, different plasma shapes, and non-linearity effects connected with refractive index of the plasma is performed. The interferometer will provide information on the line-average electron density in real-time and will be used in the plasma discharge control system. Coefficients for real-time correction will be calculated based on ray tracing using numerical methods that were tested on COMPASS tokamak.

P3 / 1117

P3.070 Design of the new electromagnetic measurement system for RFX-mod upgrade

A major modification of the RFX-mod toroidal load assembly has been decided in order to improve passive MHD control and to minimise the braking torque on the plasma, thus extending the operational space in both RFP and Tokamak configurations. With the removal of the vacuum vessel, the support structure will be modified in order to obtain a new vacuum-tight chamber and the first wall tiles will be directly in front of the passive stabilising shell inside of it, so increasing both the poloidal

cross section (by 28 mm radially) and the plasma-shell proximity.
This implies the design of a new vacuum fit electromagnetic measurement system. The spatial resolution is constrained by the number of graphite tiles, 28 (poloidal) x 72 (toroidal), which cover the inner surface of the copper shell.
The new local probes (of size 42 x 37 x 7 mm) will be installed in vacuum onto the copper shell, behind the graphite tiles, and shall operate up to a maximum temperature of 180℃ to allow for baking cycles for first wall conditioning. Because of the reduced room available, triaxial pickup probes have been designed, with the additional advantage of allowing the minimization of alignment errors.
The paper describes the detailed design of the new probe set, in particular highlighting the technological issues complying with different functional requirements: the thermo-mechanical interface with the copper shell, their protection, the electrical connection for the transmission of the signal with a minimum level of noise and the choice of materials suitable to the vacuum environment at relatively high temperature. The extensive tests carried out on probe prototypes to characterise their electromagnetic, thermal and vacuum behaviour are also reported.

P3 / 1120

P3.073 Prospects for the Hall sensors based steady-state magnetic diagnostic for future fusion power reactors

The magnetic diagnostic is essential for today's tokamaks to determine the plasma position, stability, energy content and additional parameters critical for safe operation of these devices. Conventional sensors, such as the inductive sensors, have to be supplemented by steady-state magnetic sensors in devices with long pulse capability, as is planned for the ITER reactor. In ITER, the set of 60 discrete outer vessel steady state magnetic sensors, based on bismuth Hall sensors, will be deployed. The steady state magnetic sensors are considered also for future fusion power reactors starting with DEMO device. The main difference in requirements with respect to ITER will be the higher ambient temperature at sensors location (~ 350 ℃, compared to up to 100 ℃ in ITER during the normal operation and 220 ℃ during baking) and up to 3 orders of magnitude higher lifetime neutron fluence exposure. The use of the ITER Hall sensors is not possible in DEMO due to a low melting temperature of bismuth which is 272 ℃. As a consequence, alternative sensing material solutions have to be sought.
This contribution will review the present status of the DEMO steady-state magnetic sensors based on antimony and antimony/bismuth materials. Possible ways to further increase radiation hardness of the sensor with respect to Hall sensors planned for ITER will be highlighted. A control electronics for these sensors plays a key role to achieve required accuracy of measurement in the noisy environment of fusion power reactors. Advanced techniques, such as synchronous detection and current spinning techniques is planned to be used to achieve satisfactory performance of the Hall sensor controller. The concept for the DEMO Hall sensor controller will be contemplated within this contribution. Finally, the design of the "DEMO prototype" Hall sensors, recently proposed to be tested inside ITER port plugs, will be presented.

P3 / 1122

P3.075 Design Updates of Magnet System for Korean Fusion Demonstration Reactor K-DEMO

Based on the Korean Fusion Energy Development Promotion Law was enacted in 2007, a conceptual design study for a steady-state Korean fusion demonstration reactor (K-DEMO) was initiated in 2012. One of the key components of the K-DEMO, the superconducting magnet system consists of 16 TF (Toroidal Field), 8 CS (Central Solenoid) and 12 PF (Poloidal Field) coils. All of the TF, CS and PF coil system use internally-cooled Cable-In-Conduit Conductors (CICC). By using high performance

Nb3Sn-based superconducting conductor currently being used in accelerator magnet area, the TF coil system provides a field of 7.4 T at a plasma center with a peak field of 16 T. Key features of the K-DEMO magnet system include the use of two TF coil winding packs, each of a different conductor design, to reduce the construction cost and save the space for the magnet structure material. Also, the configuration is constrained by maintenance considerations, leading to a magnet arrangement with large TF coils, which minimize the magnetic ripple, and widely-spaced PF coils to accommodate the enlarged vacuum vessel upper ports to remove the in-vessel components as large sectors. CICC connection scheme between coil systems has been developed as well. The fabrication of the test specimens for the each type of TF, PF, CS CICCs was also performed to check the fabrication feasibility. The K-DEMO magnet system has been evolved mainly for solving engineering issues. The major parameters and design of magnet system including the supporting structural analyses will be presented.

P3 / 1123

P3.0706 Qualification of ITER Poloidal-Field coil cryogenic components

The ITER Poloidal-Field (PF) magnet system is composed of six circular coils consisting of superconducting winding packs made up from a stack of Double Pancakes. Due to the large coil sizes the coils PF2, PF3, PF4 and PF5 are to be fabricated adjacent to the ITER site in a dedicated PF Coil fabrication building. The cold testing of the full coils PF2 – PF6 will be carried out in the same facility.
The manufacturing design of the PF Coils has been started and the qualification of special processes and subassemblies is now completed.
To qualify the performance of PF Coil key components, several tests at room- and cryogenic temperature were performed at the dedicated test facility CryoMaK of the Karlsruhe Institute of Technology. Components under test were PF Helium Inlet, PF Joint and PF Tail full-scale mock-ups. Mechanical fatigue testing and Helium-leak test were performed to demonstrate the soundness of the welded joints of the mock-ups after the foreseen number of operating cycles. Small size specimens of the insulation material were investigated under different mechanical test modes to study the structural integrity. Finally, PF 3X3 conductors coil section mock-ups were also provided by F4E to investigate the mechanical and electrical integrity of a winding pack.
The work summarizes the significant results demonstrating the sound performance of the provided components.

The views and opinions expressed herein do not necessarily reflect those of the ITER and F4E Organizations.

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P3.080 Development of Insulation Technology with Vacuum-pressure-impregnation (VPI) for ITER PF6 Double Pancakes

The Poloidal Field(PF) coils are one of the main sub-system of ITER magnets. The PF6 coil is being manufactured by the Institute of Plasma Physics, Chinese Academy of Sciences(ASIPP) as per the Poloidal Field coils cooperation agreement between ASIPP and Fusion for Energy(F4E).
The PF6 coil winding is constructed from nine double pancakes (DPs) which are plane cylindrical solenoids of about ∅10.2m×0.12m in size and about 29 ton in weight. The turn and pancake insulation of DPs are electrically insulated with a glass tape/polyimide interleaved multilayer composite, which should be impregnated with epoxy resin. Series DPs are being insulated in ASIPP with the vacuum-pressure-impregnation (VPI) technology, which is an effective process to remove defects such as dry spots and over rich resins in the insulation system. In this paper, the authors introduce the detailed VPI process including the insulation materials and insulation structure for PF6

DPs and present the results of the related geometry and electrical tests performed. The geometry parameters are assessed with profile tolerance, flatness as well as the parallelism by a laser tracker. meanwhile, the high voltage (HV) tests of the DPs insulation, including the direct current (DC) and alternating current (AC) tests are examined. The results of the tests are discussed versus the design requirements.

P3 / 1124

P3.077 ANALYSIS OF TORE SUPRA/WEST TOROIDAL FIELD COIL QUENCH AND THERMO-HYDRAULIC MODEL WITH SUPER-MAGNET

The Toroidal Field (TF) system of the Tore Supra/WEST tokamak comprises 18 NbTi superconducting coils, cooled by a static superfluid helium bath at 1.8 K and carrying a nominal current of 1255 A. The 19th December 2017, at the end of plasma run #52205, the current Fast Safety Discharge (FSD) was triggered after a quench of TFC-09.
A numerical model has been developed with SuperMagnet (CryoSoft). The whole TFC-09 circular coil is modelled by THEA as a single big Cable-In-Conduit Conductor (CICC) with 2028 big rectangular strands (monolithic conductors of length equal to coil average perimeter). The external quench helium relief circuit (cold and warm safety valves, rupture disk and magnetic valve with corresponding pressure set) is modelled by FLOWER.
The paper compares the calculated results and measured signals during TFC-09 quench, with focus on 3 main important parameters:
1) the Minimum Quench Energy (MQE) calculated in THEA with small initial heat deposition length (few tens of centimeters) at low field region (external leg) which is compared to the real shape and energy of neutron and gamma flux caused by highly energetic runaway electrons colliding the outboard plasma facing components. This energy is found to be in the order of few kJ.
2) the calculated expulsed helium mass flow through cold safety valve compared to the measured one. The whole initial helium mass of 34 kg is expulsed in nearly 5 s giving a mass flow of nearly 6 kg/s.
3) the helium pressure in the coil, upstream of cold safety valve, with a maximum value of nearly 9 bar.
This event confirms the criticality and the possible occurence of a so called “smooth quench” caused by small initial heat deposition length and at low field region which can be useful for other Tokamak magnets quench studies and safe operation.

P3 / 1130

P3.083 Simulation and verification of air cooling system for MITICA High Voltage Deck installation in Padova

The first ever built full-scale prototype of the ITER heating neutral beam injector is the MITICA experiment at PRIMA-NBTF, under realization in Padua, Italy.

The MITICA experiment includes many auxiliary plants; this paper is focused on the integration of the High Voltage Power Supply (-1 MV), hosted in a Faraday cage (HVD, High Voltage Deck) inside a dedicated Building at PRIMA-NBTF.

The HVD hosts the MITICA ISEPS (Ion Source Power Supply) in a two-floor cage, ground insulated, with huge dimensions (11 m x 10 m x 9.5 m, LxBxH).

Most part (around 85%) of the heat load generated by ISEPS is cooled by a deionized water dedicated circuit, while around 100 kW are rejected to the air surrounding the equipment located in the Faraday cage.

Severe cleanliness constraints are required in the Building hosting the HVD, as caused by high voltage operative conditions (the -1 MV Faraday cage is air insulated).

A dedicated air-forced ventilation system is part of HVD procurement; this system has been integrated in the Building conventional air conditioning system (part of building Contract) by studying operative scenarios with a Computational Fluid Dynamics (CFD) tool, as well explained in this paper.

Different operating scenarios were analyzed with this tool in order to optimize the Building air conditioning system, to improve the positions of air inlet and outlet nozzles, to verify the system size, to check the efficiency of heat loads removal, and finally to investigate about possible zones with unacceptable air turbulence conditions.
Functional logics of the building conditioning system have been finally implemented both considering normal and exceptional operative conditions (like the intervention of fire extinguishing system of the insulation transformer located in the same Building).

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P3.088 ITER TF coil OIS interface assembly tool

Assembling of ITER components is a major challenge especially because of the large size and weight of its components, and the high accuracy that has to be reached. As part of the Magnets Infrastructures Facility for Iter (MIFI) agreement between CEA and IO, CEA works on the assembly sequence of ITER TF coils OIS (Outer Interface Structure) composed of shear pins and bolts. These pins and bolt weight approximately 35 kg, and are located in a small space which considerably complicates their manipulation and assembling. A specific tool was developed by the CEA design office in order to achieve the assembling of these bolts and pins. This tool, inspired by tele-manipulators, uses complex kinematics and a counterweight in order to be equilibrated in any position, allowing an operator to easily manipulate a 35 kg bolt. As the assembly sequence and the accessibility are very constrained, virtual reality simulations with contact detection was necessary to define and validate the range of each joint and the volume occupied by this tool.
First the ITER context and the assembling sequence of two ITER coils are presented. Then, the concept of the tools is explained and the methodology for its dimensioning is explained. Finally, the assembly sequence of the bolts with the tool is validated thanks to VR simulation developed under Unity.

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P3.107 Heat Flux Uniformity Qualification for First Wall Panel Prototype

The ITER first wall panels are exposed directly to thermonuclear plasma and must extract heat loads of about 2 MW/m² (Normal Heat Flux) to 4.7 MW/m² (Enhanced Heat Flux). The manufacturer of the normal heat flux first wall panels shall be qualified through deep high heat flux cyclic testing campaign counting thousands of cycles within the heat flux range up to 2.5 MW/m². To ensure correct testing conditions and confirm possible damage evolution during individual cycles, the heat flux must be applied as much uniformly as technically achievable. Uniformity of the heat flux application was measured on a castellated graphite plates by using infrared cameras and set of thermocouples in defined locations and checked on flat copper mock-up via X-ray imaging. Experimental tests of the applied heat flux show promising results reaching high uniformity of the electron beam irradiation and, thus, high uniformity of applied heat flux on the surface of testing sample. Uniformly applied heat flux can be then considered as qualified and ready for its application on the first wall prototype testing campaign.

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P3.092 Acceleration Grid Power Supply Conversion System of the MITICA Neutral Beam Injector: On Site Integration Activities and Tests

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The Acceleration Grid Power Supply (AGPS) is a system devoted to supply the acceleration grids of the MITICA experiment, the full scale prototype of the ITER Neutral Beam Injector (NBI). The AGPS is a special switching power supply with demanding requirements: high rated power (about 55 MW), extremely high output voltage (-1MV dc), long duration pulses. The procurement of the AGPS is split in two parts: the low voltage conversion system, namely the AGPS-Conversion System (AGPS-CS) and the high voltage DC Generators (AGPS-DCG).
The AGPS-CS has been installed and commissioned in between late 2017 and early 2018. The design of the installation engineering has been particularly challenging. Some critical issues concerning the integration with the existing buildings and auxiliary systems were assessed during the installation period. Examples of such issues are the definition of the medium voltage cable layout and interfaces, the design of high current busbars penetrating in the building for compatibility with the prescriptions about fire propagation, the adaptation of the infrastructures to enhance the maintainability of the system. Another important matter has been the assessment of the passive filters needed to damp the voltage oscillation at the end of the cables connecting the AGPS-CS to AGPS-DCG, which were demonstrated to be necessary after the analysis of the cabling layout. Concerning the commissioning and acceptance tests on site, they were performed on a dummy load at reduced power and duty, testing one section of the system at a time, being very difficult to design and realize a test at full rated power. This paper is focused on the installation and commissioning activities, highlighting the peculiar integration aspect of this unique system, above described. Moreover, the results of the site tests are presented, showing the ability of the system to provide the required performance.

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P3.093 Detailed structural analysis of a graded TF coil winding pack for EU DEMO

The toroidal field (TF) coil system is one of the most mechanically stressed system in a tokamak. Structural integrity of the system must be maintained on the global and on the local scale, where the stress state in each conductor jacket as well as in insulation is to be within structural allowables. Solving this task head-on leads to very high computational demands. In this work a methodology for a FE detailed analysis of a graded TF coil winding pack (WP) for EU DEMO is presented.
The procedure relies on the submodeling approach. In the global model the WP is approximated by smeared material properties. The cut boundary displacements are transferred to the submodel which consists of a piece of the TF coil case with the detailed WP geometry. Operational electromagnetic loads are applied as hydrostatic pressure to the conductor jackets and the superconducting cable is omitted from the submodel for additional computational savings. The contact interaction between the case and the WP is taken into account as well. The procedure is implemented parametrically using ANSYS APDL macro language and allows automatic generation of the submodel geometry, meshing and recalculation for faster assessment of new design iterations.
A graded WP design option was analyzed in several critical locations of the coil, which were identified by an approximate, but very fast approach [1] developed earlier. A comparison with the said approach was made to verify its applicability and limitations. Convergence studies were conducted to find the required length of the submodel cut piece.
The presented procedure proved to be accurate requiring only moderate computational efforts. The parametric approach allows fast implementation of design changes into the model. The procedure can be used to speed up the assessment of future design iterations of the DEMO TF coil system.

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P3.096 The coil system preliminary design for a large-scale high-intensity magnetic field immunity test platform

Due to the high intensity stray magnetic field around the tokamak device, static magnetic field immunity test is an essential procedure to verify the reliable operation of electrical and electronic equipment nearby. For the safety and reliability concerns for the Chinese Fusion Engineering Test Reactor (CFETR) and future tokamak devices, a large-scale high-intensity static magnetic field immunity test platform will be built in the Chinese Academy of Sciences, Institute of Plasma Physics (ASIPP). This paper presents the preliminary design of the coil system which will be used to generate a uniformly distributed magnetic field in the specified test zone for the test purpose. At first, a magnetic field homogeneity analysis is performed to determine the structural parameters of the test coil which is in square solenoid type. It ensures the magnetic field homogeneity in the test zone. Then, the electrical parameters and the highest long period test capacity are approximately evaluated in accordance with the given cooling capacity of 4 MW. In addition, the design of cooling system for the test coil is performed based on an electric-thermal-fluid coupling analysis. A comparison of two different cooling water route connections between two orthogonal conductors is also conducted which results in an optimized cooling design. At last, further analysis is performed to evaluate the possible highest test capacity within the given cooling capacity. It indicates that the test platform can be used for the test of an equipment with a largest dimension up to 2.1 m at 0.5 T for a long period, 0.7 T for about 400 s and 1 T for about 90 s. It is of great value to ensure the reliable operation of electrical and electronic equipment in CFETR and future devices.

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P3.138 Upper vertical neutron camera integrity report

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The presentation is focused on approaches and results of simulations and used for loading analyses made for Upper Vertical Neutron Camera (UVNC), including spatial stress strain state, seismic analysis, electromagnetic analysis as well as the most important load combinations.
The Vertical Neutron Camera is a multichannel neutron collimator intended to measure the time resolved neutron emission profile for both DD and DT ITER plasmas.
Finite element model of the UVNC was created based on UVNC PDR stage design. Also, simplified model of the DSM was created. This model was used in thermal and thermal stress analyses to describe in a correct way interface loads received from DSM structure. Only in-vessel components of the UVNC structure were considered at this analysis stage.
The following single load cases are considered according to the System Load Specification: dead weight, coolant pressure loads, seismic loads, EM loads due to EM transients, thermal loads, interface loads.
Thermal-hydraulic analysis of the UVNC cooling system was conducted. Temperature, wall heat transfer coefficient, coolant pressure and velocity fields were obtained for the cooling channels. Wall heat transfer coefficient from CFD analysis was used as a boundary condition in the FE transient thermal analyses of normal operation and baking modes. Based on thermal analysis results, UVNC structural analysis was carried out, stress-strain state was analyzed.
Linear spectrum method was used in the seismic analysis to obtain volumetric distribution of stress and displacement in the UVNC structure due to seismic loads. Electromagnetic analysis was conducted for the most severe plasma disruption event to determine values of affecting structure forces and moments. Stress strain states due to plasma disruption events were obtained based on EM analysis results. Finally, the load combinations from System Load Specification were studied and the obtained results were analyzed using RCC-MR criteria

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P3.097 THE CONCEPTUAL DESIGN OF FAST DISCHARGE RESISTOR SYSTEM FOR LARGE FUSION DEVICE

When the quench occurrence in the operating course of the large fusion device, huge energy stored inside the magnetic load and the maximum current flow from the superconducting load can reach 100kA with maximum inductance value to be 2H that will lead an irreversible damage on the device. By dissipating the energy by means of the fast discharge resistor(FDR) system connected in series to the magnetic load, the current in the magnetic load will diverted from a short circuit into the FDR system.
The FDR system consists of the resistor modular matrix to form different resistance value with serial and parallel connection in the method of flexible coupling. Through the analysis and simulation course of the structure resist, deformation, heat, stray parameter and EMC effect, the structure adopts the zigzag design to connect the edges of resistor laminator with supporting bar using epoxy resin between each interval. The material of the resistor laminator takes the 304 stainless steel. The studdle of G10 material goes across the whole structure to resist the gravity effect. The design of the heat dissipation effect is to adopt forced air cooling from the bottom to the top with air exchange device.

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P3.101 Welding technologies applied on casings for JT-60SA toroidal field magnet

In the framework of the Broader Approach program, ENEA supplied the Toroidal Field (TF) coil casings for JT-60SA tokamak.
ENEA commissioned the manufacture of the full set of eighteen casings for the integration of the TF coils plus two additional spare casings to the company Walter Tosto (Chieti, Italy).
The casing is segmented in one outboard straight leg, an outboard curved leg and three inboard covers. The preliminary design of the casing components has been prepared under the coordination of Fusion for Energy. The detail design has been finalized involving industrial partners responsible for the subsequent integration of the coils
Contract started in 2012 with the design and fabrication of mock-ups representative of the most important cross sections of the casing complete with the relative chamfers. Detail design of the casing components was completed in 2013 and supported by qualification of welding processes and definition of manufacturing procedures. Casing production activities started in 2014 and the full procurement of the eighteen casings plus two additional spare casings was completed within August 2017.
This paper provides an overview of the welding technologies applied for the casing procurement such as the selection process, qualification activities, welding production, final checks and considerations.

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P3.102 Effect of initial microstructure on the retention and desorption of deuterium and helium from W exposed to plasma

In the process of burning fusion plasmas, plasma-facing materials such as tungsten-based materials (W) will be exposed to energetic particles of hydrogen isotopes and helium (He), high heat flux, and neutrons. In this regard, a study of accumulation of hydrogen isotopes and He in W under normal operation conditions and transit events appears necessary for assessment of safety of fusion reactor

due to the radioactivity of tritium and material performance and for the plasma fuel balance. In the present work, we focus on understanding of an effect of initial microstructure of W on the fuel and He retention. Polycrystalline bulk W (PCW) and dense nano-crystalline W coating produced by Combined Magnetron Sputtering and Ion Implantation (CMSII) technology have been exposed to deuterium (D) and He plasmas at different specimen temperatures ranged from 320 to 1250 K up to fluences of 10^26 at.m-2. We found that both the D and He retentions are higher in nano-crystalline W compared to PCW. The D retention decreases but the He retention slightly increases in both PCW and CMSII-W coating with increasing the temperature from 500 K to 1250 K. The seeding of 5-15% of He into the D plasma reduces the D retention in both PCW and CMSII-W coating but this effect is more pronounced for PCW. The nanostructured W 'fuzz' formation due to He plasma exposure at temperatures of 1000-1250 K results in the increase of the He retention in both PCW and CMSII-W coating but the threshold of the W fuzz formation is shifted to higher temperatures in the case of nanostructured coating. The additional low-temperature desorption stages below the irradiation temperature were found in W materials after fuzz formation. The physics of D and He retention and desorption and ways of their reduction are discussed.

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P3.104 Tritium permeability in polycrystalline tungsten

Nuclear fusion promises to deliver an abundant, carbon-free and clean energy source for the future. Before the realization of nuclear fusion energy, the fusion community must solve immense technological safety challenges related to tritium permeation in materials under an extreme fusion nuclear environment. Tritium behavior in materials determines two crucial safety evaluation source terms: in-vessel inventory source term and ex-vessel release term, which are used in reactor safety assessments for licensing fusion facilities. Extensive work on hydrogen and deuterium permeation behavior in fusion materials has been conducted, but only a small database is available for tritium, the radioactive fuel for future reactors, due to the cost and difficulty associated with handling tritium [1]. A gas-driven tritium permeation system that is capable of independently controlling hydrogen and tritium partial pressures was developed at Idaho National Laboratory to enhance the tritium permeability database in fusion reactor materials [2].
In this paper, we discuss tritium permeation behavior in polycrystalline tungsten (99.95% purity, 20.0 mm OD disc, 25.4-127 μm thick) in a hydrogen-tritium system (T2 partial pressure range from 10-6 to 10 Pa) and specimen temperatures up to 600 ℃ to uncover the basic scientific understanding of tritium permeation behavior in tungsten, the leading candidate plasma-facing material for DEMO.

[1] R.A. Causey, R.A. Karnesky, and C. San Marchi, "4.16 Tritium Barriers and Tritium Diffusion in Fusion Reactors", Comprehensive Nuclear Materials (2012) Chapter 4.16.
[2] M. Shimada and R.J. Pawelko, "Tritium Permeability Measurement in Hydrogen-Tritium System, Fusion Engineering and Design, 129 (2018) 134.

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P3.105 Investigation of hot spots development on metallic PFCs in the JET-ILW

Plasma Facing Components (PFC) in JET with metal ITER-like wall are subjected to high heat fluxes which can lead to damages such as beryllium melting or thermal fatigue of tungsten. The hot spots formation at the re-ionization zones due to impact of the re-ionised neutrals injected by the heating

system as well as due to RF-induced fast ion losses is recognized as a big threat due to quick surface temperature rise. To address this issue a real-time algorithm of automated hot spot detection called vessel thermal map (VTM) with help of Near-Infrared (NIR) protection imaging system was installed. Because these hot spots could trigger VTM alarms, which could cause the protection system to stop a pulse, it is important to identify the mechanisms and conditions responsible for the formation of hot spots.
To enable easy tracking of hot spots history and their evolution a new software tool Hotspot Editor was developed. It is based on the catalogue of hot spots in the main chamber and on divertor tiles detected during experimental campaign and is now routinely used at JET in support of the preparation of plasma pulses. As a result, the number of the VTM alarms was significantly reduced during the last campaigns. In this contribution, we will also provide an analysis of hots spot formation in relation to the diverse plasma parameter as well as conditions of possible disappearance of hot spots (e.g. after the cleaning up of tiles). We will also discuss the recommendations for avoiding or minimizing hot spots and VTM alarms.
Such a sophisticated analysis formation and development of hot spots becomes especially important during the preparation and execution of the experiments of the coming D-T campaign, where stationary plasmas with additional power of 40MW/5s and with tolerable wall heat loads and impurity concentration are required.

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P3.132 Fabrication and Factory Acceptance Test of ITER Sector Sub-assembly Tool

An ITER Tokamak machine with a torus shape is composed of nine units of 40◦ sectors. The sector sub-assembly tool (SSAT) is dedicated assembly tool to integrate vacuum vessel (VV) sector, VV thermal shield (VVTS) segments, VVTS port shrouds, two toroidal field coils (TFC) and various intercoil structures into 40◦ sector.

For the sub-assembly of 40◦ sector, SSAT shall have sufficient strength to support and handle heavy components up to 1200 tonnes. And fine alignment system to adjust components considering assembly tolerance should be designed and verified by appropriate means. Although the structural analysis and mock-up tests were completed, structural stability and alignment system should be verified through factory test with real component and system.

In this paper, the factory test results of the sector sub-assembly tool including VV load simulation and partial load test of VVTS outboard sector frame will be presented to verify the structural assessment and adjustment component.

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P3.108 Manufacturing of the First Wall Panel-prototype for the ITER First Wall normal heat flux design

This poster describes the main steps realized for the manufacturing of a full scale First Wall panel to ITER. This full scale prototype (FSP) is foreseen to be delivered in 2019 to F4E in order to perform high heat flux tests. The dimensions of this prototype are 1360 mm x 850 mm x 500 mm. It consists of a bi-metallic support structure made from 15-25 mm thick CuCrZr alloy embedded with 316L(N)-IG tubes and bonded to a 40-50 mm thick 316L(N)-IG backing plate with cooling channels. Moreover, the CuCrZr surface is coated with 784 Beryllium tiles.
The Technical Center of Framatome in Le Creusot used its competencies in conception, manufacturing engineering and lessons learned with the previous mocks-up to successfully manufacture and assemble this FSP. Many technologies were carried out : diffusion bonding in solid state by Hot Iso-

static Pressing (stainless steel to stainless steel, stainless steel to CuCrZr alloy), stainless steel and CuCrZr alloy machining (deep drilling, milling, etc.), welding, bending…
Currently, the FSP reaches a new stage : the entire structure is finalised and joined. The progress of the assembly which is now in its final shape will be presented. Also, the previous steps of the manufacturing will be shown. Finally, control stages : helium leak test, ultrasonic tests and specially 3D scan measurements will be described.

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P3.109 CFD simulation of the water cooling of the DEMO first wall under high heat flux exposure

The water-cooled lithium lead (WCLL) blanket is considered as one of the possible candidates for the EU DEMO blanket in the present EU fusion roadmap. One of the critical points of the first wall design is the maximal allowed thermal load of the Eurofer97 steel within the limiting temperature of 550 ℃. Therefore, the initial reactor geometrical concept of the WCLL blanket allows a heat flux load on the plasma facing surface up to about 0.75 MW/m2 only.
The effects of the design changes are evaluated using thermohydraulic three-dimensional numerical simulation. The simulation results show that the width of a blanket module and the thickness between the plasma-facing surface and the cooling channel have a significant role in the surface layer temperature gradients and limit the maximum allowable heat flux to the surface. Geometrical modifications are proposed to increase the limit heat flux under defined thermal conditions.

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P3.111 Application of RANS CFD models for helium cooled first wall with fins for heat transfer enhancement

Helium cooled First wall (FW) is being developed within the breeding blanket (BB) workpackage of the EUROfusion project as one of FW options for the European DEMO. The helium cooling system has to be adapted to high thermal loads and at the same time to achieve reasonable hydraulic parameters. Moreover, different values of the heat fluxes are expected at the DEMO FW depending on the position inside the tokamak. While the most of the FW can be cooled with bare helium ducts, which can be relatively easily characterized by available computational models and correlations, implementation of measures for heat transfer intensification at locations with higher heat fluxes is necessary. The present work deals with one of the possibilities for the heat transfer enhancement which is an application of fins on the channels surface. A RANS (Reynolds-averaged Navier-Stokes equations) CFD model was developed using ANSYS FLUENT CFD code as an option for simulation of this component. The model was benchmarked on the HETREX experimental data considering a DEMO relevant thermal hydraulic conditions and then applied on the full scale computational model of one segment of the Helium Cooled Lithium Lead (HCLL) BB FW. Results of the comparison with the experimental data along with evaluation of the model applicability are reported in the paper. Evaluation of efficiency of selected fins layout in terms of heat transfer for the HCLL conditions and assuming peaking heat fluxes was also performed. Apart from this, the main challenges and recommendations for the computational mesh preparation and numerical solver settings learned from the model development phase are also mentioned. Even though the RANS models cannot affect all physical phenomena, they have some advantages over the other computational methods and can be a useful tool for specific kinds of design and safety studies of the helium cooled FW.

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P3.113 ENEA Ultrasound Test of Plasma Facing Units

The Plasma Facing Units are the components of the ITER's divertor target exposed to the plasma. PFUs are cooling pipes made of copper covered by tungsten monoblocks as armour.
The non-destructive ultrasonic control is the simplest and most economical test for PFU control. It has also proved to be extremely reliable and accurate in identifying and sizing defects.
ENEA has been working on improving the scanning technique of these components for many years, obtaining increasingly accurate and easy to interpret results. The first tests were performed on small mock-ups and, due to the geometry of the component, tests were performed from inside the tube. The first problem to solve was the reduced pipe thickness and the need to direct the ultrasound beam orthogonally to the tube surface. The choice of the correct probe characteristics and its particular shape solved this problem. Subsequently, the problem of a scan along a curved tube had to be addressed. Then, a flexible tube, at the end of which the probe was placed in a special probe holder able to accompany the curvature, was achieved. The definition of more stringent acceptance criteria for these components has required to increase the scanning resolution. Working on automatic system and data analysis software, we have come to detecting and sizing defects up to 0.5 mm. As sample sizes increase up to full-scale prototype for pre-production, the problem has become to speed up scanning without losing resolution. At this point, the scan along generatrix was discarded to pass to an automatic system that proceeds with a helical path. In this work, the system created and patented in ENEA laboratories for scanning of full-scale Inner Vertical Target PFU prototypes of ITER is illustrated. Test results in terms of resolution and defects sizing are shown.

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P3.115 Tungsten sources and transport during Metal Rings Campaign in DIII-D*

The Metal Rings Campaign in DIII-D allowed for studies of tungsten sourcing and transport from poloidally localized, isotopically distinct surfaces in a low-Z background. Two 5 cm wide toroidal rings of W-coated tile inserts were installed in the lower divertor of DIII-D. The outboard (shelf) ring was coated with isotopically enriched W-182; the inboard (floor) ring used a natural W coating. Two types of collector probes were used in the divertor and near outboard mid-plane to study short- and long-range transport of W. The divertor probe installed on DiMES manipulator was located 1 cm radially outboard of the shelf ring. The mid-plane collector probe was inserted in the scrape-off layer (SOL) plasmas. In reproducible L-mode plasma discharges with Outer Strike Point (OSP) on the shelf ring, a low near-surface steady-state W coverage formed on the part of the divertor probe undergoing net erosion, while in the region of net C deposition W kept accumulating with time in the deposits. In the same experiment, measured deposition of W on the mid-plane collector probes showed more W collected on the side facing away from the lower divertor, which was explained by OEDGE modelling suggesting W accumulation at the plasma crown, driven by ion temperature parallel gradient force. In another experiment with ELMy H-mode discharges, the OSP was positioned on the floor ring while the shelf ring was in SOL. Isotopic analysis of W collected by mid-plane probes showed that for discharges with smaller frequent ELMs the floor W source was dominant, while with larger, less frequent ELMs the floor and shelf sources were comparable. Arcing observed on the shelf ring via fast filtered imaging may be also a significant contributor to the SOL W source during large ELMs.

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P3.117 Permeation properties of deuterium in tungsten exposed to D-He mixture plasma

Divertors are responsible for removing the exhaust helium ash generated after fusion in a magnetic fusion reactor. Tungsten (W) was selected as the plasma facing material in the ITER divertor region because of its high melting temperature and thermal conductivity and low sputtering erosion yield. Therefore, it is crucial to understand the behavior of hydrogen isotopes in W contained in the diverter wall material. Effects of He under the D–He mixture plasma on the retention and permeation properties in W of the divertor wall material are investigated. Experiments have been carried out in a linear plasma simulator TPD-Sheet IV. D and He mixture plasma is used, He gas is added at 10 sccm, and ITER-grade W is used as the sample. The permeation property of D in W is investigated using a Ti plate as the D storage material, which is mounted behind W. The retention amount of D in samples is determined by the signals of the thermal desorption spectroscopy (TDS), respectively. The ion density in the D-He mixture plasma measure by the omegatron mass analyzer. The retention and permeation properties in tungsten investigate as a function of the incident flux. The plasma incident fluence is 4.4× 10^20 m^2 for all the plasma incident fluxes. As the plasma incident flux increases, the retention amount of D in W decreases in the D–He mixture plasma irradiation. On the other hand, the retention amount of D in Ti greatly increases. The effects of the D–He mixture plasma on the deuterium retention properties are different depending on plasma incident flux.

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P3.119 FABRICATION ROUTE OF THE ANSALDO-ENEA ITER INNER VERTICAL TARGET DIVERTOR FULL SCALE PROTOTYPE

The scope of contract F4E-OPE-138 Lot 1, assigned to Ansaldo Nucleare S.p.A (ANN) by Fusion for Energy, the EU-Domestic Agency, is the fabrication and qualification of a representative full scale prototype of the International Thermonuclear Experimental Reactor (ITER) divertor inner vertical target which procurement falls under the EU responsibility.
ENEA, as major partner of the contract activities, was in charge of the plasma facing units (PFU) fabrication by means of Hot Radial Pressing (HRP) process and responsible for the CuCrZr/Cu joined interfaces non-destructive examination which was performed by ultrasonic water gap technique by means of a dedicated equipment designed by ENEA itself.
In parallel the other main fabrication steps (W monoblock preparation, brazing of the XM-19 support, CAM machining of the steel support structure and test frame, welding, dimensional checks, etc.) were completed by ANN and its industrial partners based on the F4E technical requirements.
The feasibility of the prototype assembly by integrating the PFUs onto the steel support structure has been demonstrated and the technical capability to reach the tight geometrical tolerances required for the plasma surface is being checked and will be reported.
The high heat flux thermal fatigue testing that will take place during 2018 on the ITER Divertor Test Facility at Efremov Institute in Saint-Petersburg on 8 full-W PFUs installed on the testing frame, is aimed at finally demonstrating the PFUs thermal capability to remove up to 20 MW/m2 during transient events of 10 seconds, while normal operation heat flux on the bottom segment of the divertor is up to 10 MW/m2 steady state.
The reliability level reached by the HRP joining process and the industrial consolidation of the fabrication route, paves the way with confidence to a forthcoming series production of the ITER full-W divertor. The paper reports the full scale prototype main production achievements.

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P3.122 Stability of Be-based coatings under annealing up to 1273

K

Erosion, co-deposition of impurities and heat load effects leads to compound formation in PFC enhancing delamination mechanisms with re-emission of dust particles detrimental to the plasma stability of fusion devices. Beryllium (Be) and tungsten (W) PFC were used in the first wall and divertor in JET, and carbon (C) has achieved new relevance as impurity in the same reactor. Earlier investigations of compound formation were typically performed in the absence of oxygen (O). Nevertheless, the formation of significant BeO contents is also detrimental to the mechanical integrity of mixed layers and should be considered in face of possible accidental O gathering in large reactors as in ITER.
We investigated the formation of compounds and the mechanical integrity of coatings within the Be-C-O and Be-W-O systems under annealing up to 1273 K with atmospheric base pressures close to 1 x 10-7 mbar, making use of electron microscopy, ion beam, X-ray diffraction and nano-indentation techniques. In the Be-C-O system the formation of Be2C and BeO is enhanced at temperatures higher than 873 K, leading to the huge delamination of C thin coatings deposited on Be plates and to a smooth delamination behaviour of metallic Be deposited on graphite.
The Be-W-O system was initially investigated by depositing W and Be coating on Be and W plates, respectively, being the formation of BeO, Be2W and mainly Be12W identified at 1073 K. Remarkably, the growth of berylides promotes the mechanical stability in the coatings. Even the recovery of localized delamination events in Be films occurred at 973 K was observed at 1073 K.
In opposition to the case of Be2W and Be12W, the growth of Be22W was induced by annealing Be:W layers deposited on Be plates at 1173 and 1273 K, leading to a huge mechanical failure in both films.

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P3.124 W2C reinforced tungsten: thermo-mechanical and microstructural properties

Lately, tungsten has gained considerable attention of the fusion scientific community due to its performance at high temperatures. On the other hand, tungsten is affected with a serious reduction of strength at elevated temperatures, latter being one of the main drawbacks of its usefulness as a plasma facing material in fusion reactors.1 Therefore, the main aim of this work has been to improve the material properties to sustain plasma-facing conditions when being installed in divertor as a monoblock, especially to be able to resist high thermal loadings during operation. Among the available options, we selected reinforcement of tungsten by incorporation of carbide nanoparticles, wherein the reinforcement should not chemically react with the matrix. In this respect, W2C nanoparticles offer an interesting option.
WC nanoparticles and graphene were used as a carbon source. Both used precursors resulted in the formation of W-W2C composites, due to high-temperature reaction with W powder during sintering. Using field assisted sintering technique (FAST, 1900℃, 60 MPa, 5 min), only two phases were detected in the sintered composite: cubic W and hexagonal W2C, which implies complete reaction regardless of the carbon precursor used. In addition to thorough microstructural and phase analysis, thermo-mechanical properties at room and elevated temperature of the as-sintered and samples aged at 1600 ℃ for 24 h were evaluated. The results confirm that presence of small W2C grains enhance densification of tungsten and inhibit the tungsten grain growth at high temperatures up to at least 1300 ℃. Results of thermo-mechanical measurements indicate that small amount (i.e. 5 wt %) of W2C reinforcements improve mechanical properties of the material, while the thermal conductivity of the composite does not drop below 100 W/m K at 1000 ℃. The obtained results suggested WC nanoparticles as the most convenient precursor for the formation of W2C in tungsten matrix.

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P3.127 On the alignment of the Scraper Elements at Wendelstein 7-X

Wendelstein 7-X (W7-X) is equipped with ten symmetric arranged divertor units consisting of horizontal and vertical targets each. In the current completion phase, Scraper Elements (SE) have been installed in front of two out of ten divertor units to protect the gap between the horizontal and vertical targets (pumping gap) from thermal overload out of the plasma. During the next plasma operation, investigations of the principal behavior of the SE should clarify their influence on the pumping rate as well as their capability of thermal protection. It is expected that the right functionality as well as the risk of a thermal overload of the SE depend sensitive on the geometric accuracy of SE shape and alignment.
The paper reports on the SE assembly with special focus on the geometric accuracies in shape and alignment.
After delivery of the SE (dimension: approx. 700 x 400 x 300 mm^3) an incoming inspection has confirmed the high accuracy of the graphite machining. The Best-Fit of the as-built surface shows maximum deviations lower than 0.6mm compared to the CAD. Local steps in the graphite surface are much smaller.
From plasma physical reasons, each SE has to be aligned locally to its corresponding divertor unit. This makes it necessary to adapt the reference co-ordinates of each SE according to the as-built position of the divertor unit measured in the last completion phase of W7-X. In addition, the expected environmental conditions during assembly have been taken into account for calculation of new reference co-ordinates.
The assembly procedure of the SE is presented in the paper including the measurement processes and the datum systems used to orientate the measurement tools. Finally, the alignment of the SE has been performed within an accuracy of less than 1.5mm, which fulfill the positioning requirements.

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P3.128 Modeling study on divertor plasma by impurity seeding at different locations with SOLPS-ITER

With the aims of high performance plasma toward ITER and even a fusion reactor, heat exhaust would be a serious problem for HL-2M. In this work, impurity seeding is considered to solve heat exhaust problem by radiative divertor. SOLPS-ITER simulations are performed for Ne and Ar impurities from three seeding locations (lower dome, inner target and outer target) with the standard lower single null configuration.
Simulation results demonstrate the similar high efficiency in reducing heat power of Ne and Ar, and impurity seeding is an efficient method to assure the power load to target plates at an acceptable level even at high heating power (20 MW) during discharges. The efficiency of radiative power exhaust are compared, and Ar can reach higher radiative fraction and it radiates strongly in divertor region as well as inside the separatrix. In addition, modeling shows that total radiation fraction increases with increasing seeding rate of Ne, with most of the radiation coming from divertor region. Simultaneously, the impact of seeding location on divertor plasma is identified, and explained by analyzing impurity distribution and power radiation. Furthermore, the impacts of impurity radiation on the ratio of radiated power to the power into the scrape-off layer and on the effective charge number are analyzed with different seeding rates and locations. Moreover, the divertor plasma is found to have a transition from conduction-limited regime to complete or partial detachment with the increasing of seeding rate. In particular, the peak heat flux and electron temperature on targets are both move away from the strike point with increasing throughput of impurities.
For controlling plasma-surface interactions, the results indicate that Ne is preferential to be used as the radiator. However, the level of Ne seeding rate and seeding location should be modulated to maintain the effective charge number at acceptable values.

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P3.130 Evaluation of feasibility and costs of alternative magnetic divertor configurations for DEMO

The European roadmap to the realisation of fusion energy has identified a number of technical challenges and defined eight different missions to face them. Mission 2 'Heat-exhaust systems' addresses the challenge of reducing the heat load on the divertor targets. Part of this mission is an assessment of several alternatives to the conventional divertor configuration, including 'Advanced Magnetic Configurations' such as the double null, SnowFlake, X and Super-X divertors.
In this paper, starting from the geometrical description of a conventional European DEMO scenario with an aspect ratio of 3.1 and a reference single null configuration, we investigate the feasibility and the costs of the alternative divertor configurations (ADC) on DEMO. We have developed DEMO descriptions for ADC optimizing the plasma shape, the machine geometry (first wall, divertor structure, vessel and TF coil shells) and the PF coil system. The magnetic configurations, designed at the same plasma current as the reference single null, feature the main characteristic of each alternative divertor concept with a constraint on the plasma-wall distance and on the plasma elongation. The feasibility of the configurations is evaluated in terms of maximum vertical force and current density on the PF coils at the start of the current flat top (SOF) and at the end of the flat top (EOF). A preliminary vertical stability analysis and a shape sensitivity analysis in case of VDE and disturbances (ELMs and MDs) have been also performed. The costs of the alternative configurations are quantified in terms of fusion power parameters, such as plasma volume and maximum flux swing at the flat top, and costs of the PF and TF coil systems, such as total current request on the PF coils and the ratio between the TF coil volume and the plasma volume.

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P3.133 Neutronics analysis for ITER UP18 preliminary design

The nuclear heat and the shut-down dose rate (SDDR) in the ITER upper port 18 (UP18) was estimated to provide the nuclear heat load for the structural analysis of UP18 and to provide the basis for the further SDDR mitigation strategy of UP18. The UP18 MCNP model has been developed based on the actual CAD model, which was integrated into C-model, the global MCNP model for ITER. While ITER UP18 accommodates three tenant systems of VUV(Vacuum Ultra-Violet)-edge Spectrometer, Neutron Activation System (NAS), and Upper Vertical Neutron Camera (UVNC), detailed models of VUV-edge and NAS were included in the UP18 MCNP model and UVNC was represented as conceptual stainless steel boxes due to lack of detailed model. For SDDR estimation, overall SDDR at the UP18 interspace were estimated as well as the SDDR contributions from tenant systems. The SDDR result includes the crosstalk from neighboring ports which are filled with general port models included in C-model. Several SDDR mitigation options were also evaluated such as to use low Co contents material, to place B4C shielding at the rear box of port plug and to use lead shielding at the interspace floor. The results indicate that evaluated options can contribute to the mitigation of SDDR in interspace, and especially to place B4C shielding at the rear box region of port plug structure can effectively mitigate the SDDR behind the closure plate.

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P3.134 Validation of cyanide copper electrodeposited threaded insert for anti-seizing in tokamak vacuum vessel

Vacuum vessel which is the first confinement barrier of Tokamak fusion reactor should have numer-

ous interfaces such as Blanket, In-vessel-coils, etc. Those interface components should be assembled by fastening of special shape of bolts to the threaded holes in the Vacuum vessel with threaded inserts and to be disassembled for maintenance during Tokamak operation. Threaded connection between Stainless steel housing and stainless steel insert will definitely cause difficulties during installation due to rather large friction coefficient between the surfaces because of fastener loads for the in-vessel components are high and require having identical material for bolts and nuts. In order to solve this issue, threaded surfaces of inserts need to be coated by Cyanide copper for anti-seizing. This coating can aid assembly/disassembly by preventing seizing of parts when low torqueing magnitude or repeatability is not critical. During the Tokamak operations, the coating will prevents seizing from high pressure on the contact surfaces of any component under ultrahigh vacuum (1×10-7 Pa m3s-1 (air equivalent) and high temperature (100 ~ 250℃). The consequences of failure of the coating will be the inability to perform maintenance activities without resorting to recovery scenarios such as bolt drilling. In this study, outgassing, adhesion and porosity and pin-on-disk reciprocating test for Cyanide copper electrodeposited surface on the 316L(N) austenic stainless steel test coupons had been performed in order to validate a high level of confidence in the performance of Cyanide copper electrodeposited threaded insert for anti-seizing in ultrahigh vacuum condition. Consequently, the validation tests was successfully performed and Cyanide copper electrodeposited threaded insert can be applied to the Tokamak vacuum vessel.

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P3.135 Preliminary electromagnetic loads calculation for the divertor of K-DEMO

The variation of plasma current and magnetic fields generated by superconducting magnet coils causes electromagnetic (EM) loads especially during the abrupt plasma current changes such as major disruption, the vertical displacement event (VDE) of plasma, and the fast discharge. The EM loads are one of the most important external loads for in-vessel components like blanket and divertor modules. The aim of this study is to estimate the EM loads of the K-DEMO divertor module under the major disruption, by using the commercial finite element software, ANSYS Mechanical/Emag. The superconducting magnet system of K-DEMO includes 16 toroidal field (TF) coils, 8 central solenoid (CS) coils, and 12 poloidal field (PF) coils. The magnetic field at the plasma center is about 7.4 T and the peak field is as high as ~16 T. The conceptual model of the K-DEMO divertor module including outboard and inboard targets, dome, and the cassette body with connecting supports was developed. The 22.5° model applying symmetric boundary condition on cutting surfaces was employed to calculate EM loads efficiently. The reduced activation ferritic martensitic (RAFM) steel has been considered as the structural material of in-vessel components. Since RAFM steel is a ferromagnetic material, Maxwell force caused by the magnetization of RAFM steel is also estimated.

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P3.140 Assessment on DEMO Water Cooled Lithium Lead alternative design configuration

One of the most critical components in the design of DEMO Power Plant is the Breeding Blanket (BB). Currently, four candidates are investigated as options for DEMO. One of these is the Water Coolant Lithium Lead (WCLL) Breeding Blanket (BB). During the previous years a conceptual design of WCLL BB has been developed. At the current state some open issues related to the manufacturability and design of the WCLL BB in 2016 configuration are still under evaluation. Since DEMO project is still in the pre-concept phase, the design team of WCLL BB decided to investigate, in parallel with the current studies, an alternative design of the WCLL BB internal structure, all that according to the Systems Engineering Approach. As such as in 2016 configuration, the alternative design is based

on the single module approach. The main drivers in developing the alternative design consisted on reducing the complexity of the internal structure itself and increasing the performances of the single module in terms of neutron shielding, tritium self-sufficiency, heat extraction and transportation to the primary heat transfer system. All that, taking into account the lesson learned by the studies carried out in the last three years. In particular, the segment has been conceived as a box in which the lithium lead flows directly inside the module. The rear part of the module is entirely dedicated to the cooling water manifold. A first 3d model of the alternative design has been provided, structural analyses have been carried out to optimize the internal structure against an over pressurization scenario. The results and the optimized design of the alternative WCLL BB design are here presented and discussed.

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P3.142 Maintenance Logistics for IFMIF-DONES

The DEMO Oriented Neutrons Source (DONES) is the dedicated facility for testing and enabling of the qualification for different materials to be utilized in the future fusion reactor DEMO. The neutron irradiation damages not only the material samples to be tested but also impacts the plant hardware in and around the Test Cell.
For that reason, preventive and predictive maintenance activities must be scheduled regularly to avoid critical failures during the neutron irradiation tests. Scores of parts and components have to be replaced, which causes a dense and complex flow of irradiated and non-irradiated materials.
In order to keep the availability high, the overarching goal is a short as possible downtime.
The Flow of Material has to be carefully analyzed considering all related boundary conditions of parts and components like sizes, shapes, weights and etc. of parts and components with the aim for an optimum plant layout. Next the transportation processes have to be modelled and subsequently can be simulated in order to find the optimum arrangement of transportation processes. Taking into account the previous results a maintenance management plan can be designed comprising amongst others an optimum warehouse location and spare part policy. The latter helps e.g. avoiding stock-outs by optimizing aspects of safety stock level, reorder points, replenishment processes and etc.
These processes have to be tackled in an integrated manner and may lead to an optimum Availability-to-Cost-Ratio from the logistics point of view.
In this paper an overview on the different methods of logistics optimization, the individual approaches and goals as well as on their interdependencies is given. Input and output data, boundary conditions, description of the process of integration, and etc. complement the paper.

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P3.143 Design of inboard first wall for the initial operation phase of JT-60SA

This article introduces overview of inboard first wall of the JT-60SA device, especially for the initial operation phase including the first plasma. The objective of the inboard first wall is to protect magnetic sensors from plasma. There is no cooling water for in-vessel components in the initial operation phase of JT-60SA, and it will be installed in the later phase. Graphite armour tiles are employed. The first wall should withstand forces such as magnetic forces due to eddy and halo currents, acceleration due to plasma disruption and earth quake (5G in the three directions at the maximum), and heat loads such as radiation from plasma, limiter configuration and ECRH direct irradiation.

To address these requirements, the inboard first wall was designed, which consists of graphite tiles, stainless pedestals and support bases. The support bases are welded on the surface of the inboard vacuum vessel. The stainless pedestals are bolted on the support bases. Two graphite tiles are bolted

on each support base. Shed roof shape of graphite tiles is employed to avoid heat concentration at side face of the tiles and to mitigate heat load during limiter configuration. Thermal structural numerical analyses are carried out, and it was confirmed that the stress is below the design stress against the predicted heat loads. It was also confirmed that the stress is below the limit against the predicted forces. Taking into account these analyses, it was confirmed the design of the inboard first wall for the initial operation phase was reasonable.

The inboard first wall in the later phase was also designed. The non-cooled stainless pedestals will be replaced by active-cooled copper heat sinks. The graphite tiles and the support bases will be re-used.

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P3.144 Remote handling tools for hydraulic connection of divertor cassette in JT-60SA

This article introduces remote handling tools for hydraulic connection of Divertor Cassette in JT-60SA, especially for cutting and aligning tools for re-welding accessing from inside of the cooling pipe. Remote handling system is necessary for the maintenance and repair of the divertor cassette in JT-60SA. Because the space around the cooling pipe connected with the divertor cassette is very limited, the cooling pipe is to be remotely cut and welded from inside for the maintenance. The outer diameter, thickness and material of the cooling pipe are 59.8 mm, 2.8 mm and SUS316L, respectively. The remote pipe cutting tool head equips a disk-shaped cutter blade and rollers which are subjected to the reaction force. The tool pushes out the cutter blade by decreasing the distance between two cams. The tool cuts a cooling pipe by both pushing out the cutter blade and rotating the tool head itself. The roller holder is not pushed out anymore after touching the inner wall of the pipe. In other words, only cutter blade is pushed out after bringing the tool axis into the pipe axis. Technical issues are longer life operations of disk-shaped cutter blade and slide members concerning cutter and rollers. In order to solve these issues, more than 5 kinds of rollers and cutter blades were assembled. In addition, the most optimum conditions in the pipe cutting operation were also investigated. In primary results, more than 5 times long life of cutter blade and rollers without any maintenance operations was realized.
Before welding the hydraulic connection between Divertor Cassette and vacuum vessel, aligning of the groove is required. Divertor Cassette equips bellows nearby the hydraulic connection. A prototype of the aligning tool, which aligns the hydraulic connection against the bellows, was manufactured.

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P3.146 Rescue and recovery studies for the DEMO blanket transporter

RACE has been developing a concept design for the remote maintenance system for the EUROfusion DEMO powerplant. Within the DEMO tokamak, tritium breeding blankets will require periodic replacement which is currently designed to utilize the upper vertical ports at the top of the vacuum vessel. This operation will be challenging due to the scale of the blankets (~10m tall, up to 80 tonnes). The blanket transporter has previously been presented as a key high technical risk system for the blanket replacement process. The concept has subsequently been reviewed by several independent industrial experts who highlighted break-down rescue and recovery as significant unaddressed risk. Rescue and recovery is often overlooked in the early stages of concept development, resulting in expensive, compromised solutions having to be incorporated later to comply with regulatory requirements. Addressing this issue early provides more scope for design solutions, but inevitably increases the complexity of the concept design.

This paper outlines the process used to identify the key hazards resulting from failure scenarios and the improvements made to the concept design to mitigate them to enable successful recovery and/or rescue. Possible failure modes were identified via input from industrial experts, completion of a Design Failure Mode, Effects and Criticality Analysis (DFMECA) and a Hazard and Operability studies (HAZOP). An Analytic Hierarchal Process was used to create a prioritised list of key failure scenarios, comparing the likelihood and severity of each. Key design changes for each scenario were identified, reviewed and, where possible, consolidated into a single solution to produce an efficient solution. Mitigation of the failure scenarios were achieved by providing secondary load paths in the main structure, redundancy in key components (such as motors) and the inclusion of additional features that allowed secondary recovery equipment to engage to assist recovery.

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P3.150 Design and Instrumentation of a Divertor Manipulator at Wendelstein 7-X

Steady-state and long pulse exposure of plasma-facing materials in reactor-relevant conditions are an integral step towards the qualification of next-step materials with respect to erosion, fuel retention and morphology changes in view of reactor applications.
W7-X will allow plasma operation of up to 30 minutes in its second operation phase (OP2) and thus provides an ideal framework for the qualification of materials in magnetically confined plasma conditions. A divertor manipulator system (DIMS) for W7-X with high heat load capability will follow the functionalities and design of the limiter lock system at TEXTOR[1] and DIM-II[2] at AUG, allowing dedicated tests of materials without necessary breaking of the vacuum for tile extraction.

Positioned as an extension of one the five lower horizontal divertor target plates, the manipulator head interacts with the scrape-off layer. Exposing material samples and components aims at study and qualification as well as determination of local plasma properties like electron temperature, density, flows etc. Furthermore, in-situ observation of the manipulator by adoption of the endoscope system equipped with multiple cameras and spectrometers provides surface information for intra- and inter-plasma phases. Complementary, a laser-based diagnostic system is proposed to study in-situ material composition and fuel retention.

Design and capabilities of the midplane manipulator drive train and control system[3] are developed further to accommodate the larger probe head size (35cm poloidally, 7cm toroidally) and weight(>25kg) foreseen here. The exposure duration of an intermediate, uncooled, probe-head design will be assessed by FEM simulations for different stroke depths and magnetic configurations. The sample holder design must allow rapid probe exchanges on a user-facility-type, while providing enough cooling abilities for the steady-state operation. Furthermore, the implementation of probes, other embedded diagnostics and a gas injection system will be detailed.

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P3.148 Towards an optimized maintenance of Tokamak devices tools and methods adopted for WEST

Tokamaks, as complex technical devices, need regular maintenances to insure optimal operational conditions. The major 2012-2016 shutdown, dedicated to the upgrade of Tore Supra, was the opportunity to engage important maintenance actions, preparing the restart and insuring the optimal

sustainability of the future subsystems of the WEST tokamak. An overall maintenance plan, based on a risk analysis and the use of the criticality matrix inspirited by the ITER RAMI approach, was established.

Following this long shutdown, the WEST tokamak is now facing a new ten years operation period. The larger access for international partners requires reaching the adequate reliability and availability. Having a direct impact on the scientific program execution, but also representing more than half of the hardware financial effort of the overall operation budget, maintenance activities evolved in 2017, leading to an evolution of methods, tools and organization.

From the maintenance point of view, a tokamak is a common industrial device, where an availability level is targeted, not to meet financial or industrial objectives, but to support the experimental program requirements. Consequently, standard maintenance tools and methods can be adopted. Management of technical activities had been adapted to clearly distinguish operation, curative and preventive maintenances, and upgrade projects, all these activities being not driven by the same objectives and constraints. For each WEST technical subsystems, a maintenance plan review is annually organized, focusing on current engaged activities and on the forecast maintenance actions. Working through this annual plan, all the technical incidents, notably occurring during experimental sessions, are monitored.

The method used for such analysis leading to an optimized balance between curative and preventive maintenance, and the development of conditional maintenance using continuous monitoring will be described. The evolution of the criticality versus the resources engaged will be highlighted for each major subsystems of WEST.

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P3.082 Thermal-Hydraulic and Quench Analysis of the DEMO CS coil

The Central Solenoid (CS) coil of the European DEMO tokamak will consist of five modules, namely CSU3, CSU2, CS1, CSL2 and CSL3, located vertically one above the other. The central CS1 module will be subjected to the most demanding operating conditions (the highest magnetic field and mechanical loads). The design concept of the CS1 winding pack with superconductor and stainless steel grading is developed by EPFL-SPC, PSI Villigen. The proposed design is based on 10 layer-wound sub-coils using HTS, wind & react Nb3Sn and NbTi conductors in the high, medium and low field sections, respectively. The proposed design undergoes comprehensive electromagnetic, mechanical and thermal-hydraulic analyses aimed at verification if it fulfils the design performance criteria. Our present work is focused on the quench analysis of all the CS1 conductors, aimed at the assessment of the maximum hot spot temperature. The analysis, based on the most recent iteration of the design proposed in 2017, is performed using the THEA CryoSoft code. We take into account the realistic magnetic field distribution along each conductor, heat transfer between neighbouring turns, and heat generation due to AC losses. A special attention is given to establishing the reliable quench simulation procedure for innovative HTS conductors. The obtained results should provide information for further improvement and optimization of the winding pack design.

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P3.156 Mechanical design of the hydraulic lift and rotation drive system for a DIII-D beamline

Neutral beams are one of the methods to inject power into a tokamak for plasma heating. The DIII-D tokamak has four neutral beam injectors with two ion sources each, located at toroidal angles

of 30°, 150°, 210°, and 330°. As originally installed, each could inject up to 5 MW of neutral beam power in the co-injection orientation (nearly parallel to the plasma current). One of the systems, the 210° beamline, was modified in 2005 – 2006 to on-axis counter-injection. This beamline is now being designed to be capable of either co- or counter-injection and will be injected at 18.2 degrees from horizontal. The beamline will always inject off-axis into the plasma, but must be horizontal for installation and major maintenance. A system capable of precise lifting and rotation is required.
A carriage has been designed that will carry the 210° beamline, and many of its associated systems to facilitate beamline lift and rotation. The carriage sits on a custom platform that supports high accuracy curved rails attached to it. These rails are placed in the front and rear of the carriage and are concentric with the beamline rotation axis. The carriage is fastened to the rails with precision roller blocks. The beamline will be capable of rotating 40 degrees, between 19.5 degrees in the counter-injection direction and 20.5 degrees in the co-injection direction. The power to rotate this carriage is via two, on board, electronically synchronized epicycloidal reduction drives. The gear reduction allows precise rotation with small motors.
The rear end of the beamline is raised and lowered with hydraulic rams which travel with the carriage.
A description of the design and fabrication of the lift & rotation system will be presented.
This work is supported by the US DOE under DE-FC02-04ER546981 and DE-AC02-0H114662.

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P3.158 Preliminary integration of reflectometry diagnostic inside the HCLL breeding blanket module for the EU DEMO

An initial conceptual study of integration of reflectometry diagnostics in the European DEMO has been carried out in the previous years within the EUROfusion project. This study considered antennas and waveguides incorporated in a full poloidal section attached to the Helium-cooled Lithium Lead (HCLL) breeding blanket segments. However, this concept of a diagnostics slim cassette would reduce the volume of the breeding blanket reducing in consequence the tritium breeding ratio (TBR) and it would require active cooling system of the cassette and its first wall connected to the HCLL helium cooling. Therefore, an alternative solution based on integration of reflectometry antennas and waveguides directly in the HCLL Breeding Blanket modules was proposed.
This paper presents preliminary integration of reflectometry antennas and waveguides inside the equatorial module of the HCLL Advanced Plus concept of the EU DEMO Breeding Blanket. The main aim is to insert the antennas and waveguides in the way not to change significantly the design of the module and not to alter the performance of the HCLL module. The integration of antennas and wave guides requires local modifications of the HCLL module, in particular at the level of the first wall cooling channels, horizontal stiffening plugs, helium and lead-lithium collectors at the back of the module, and in the back supporting structure (BSS). The impact on the module performance (cooling, thermal-hydraulic, thermo-mechanic) will be also evaluated.

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P3.160 Investigation on cooling performance of WCLL BB First Wall for EU DEMO

The breeding blanket First Wall is the first boundary separating the fusion plasma and its energetic particles from the rest of the Tokamak. In DEMO reactor, the First Wall integrated in the blanket is in charge of 1) removing the surface heat load connected with the charged particles and the volumetric power density arising from plasma; 2) ensuring the structural integrity of the blanket, thus maintaining the physical boundary of the radioactive breeder material in operating and accident conditions; 3) delivering the coolant to the First Wall Primary Heat Transfer System at design

temperature for an efficient power conversion into electricity. In the frame of EUROFUSION Work Package Breeding Blanket, the cooling performances of Water Coolant Lithium Lead Breeding Blanket First Wall are investigated using a CFD approach. The heat load specifications are reviewed and different poloidal positions have been studied, i.e. the poloidal position characterized by the maximum heat load (surface radiation and charged particles), the outboard and inboard equatorial positions. In the analyses the reference geometry is compared with a different cooling scheme, with a different channel geometry and with different radial positions. The configuration with two passes has better cooling performances and still some margin, being the coolant velocity lower than the limit (i.e. 7m/s). Compared with the square channels, the circular ones penalize the temperature field in the Eurofer, but the presence of steam in the subcooled nucleate boiling region is more limited. Moreover, circular channels may imply enhanced reliability of the system and manufacturing advantages, because the absence of joints in direct contact with the fluid domain. The results have demonstrated that the water-cooled First Wall is effective in keeping the maximum temperature of the solid structure below the Eurofer limit, as well as ensuring the design coolant temperature at the channel outlet (i.e. 328℃).

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P3.161 Investigating Ceramic Breeder Pebble Beds Thermomechanics Evolution Using Novel Pressure Mapping Technology

Ceramic breeder pebble beds undergo complex thermomechanical interactions during blanket operation due to stress build-up and relaxation under the effects of confined thermal expansion, thermal cycling, and creep. Understanding the evolution of such processes can aid in guiding blanket design, breeder materials developments, predicting performance and possible failure modes. This study introduces experimental techniques that enable us to create an environment in which the pebble bed stresses are organically self-generated as a result of relevant temperature gradients and magnitudes that allows us to observe the combined thermomechanical interaction effects on contact pressure rise and fall, as well as temperature fluctuations. A novel non-intrusive in-situ tactile pressure sensing technology is used to generate real-time contact pressure maps that reveal the spatial and temporal stress evolution with emphasis on understanding the roles that each of the thermomechanical forces play in dictating the pebble bed's equilibrium operating conditions. Two types of bed-wall contact pressure drop were recorded: (1) within the subsequent cycles due to pebbles irreversible rearrangements, and (2) within the cycle itself as a result of creep/stress-relaxation. As a result, by the fourth cycle, the contact pressure drops to negligible values. Additionally, the study revealed that the use of bed thermal conductivity at nominal packing configuration needs to be reevaluated since the thermal conductivity can vary with the pebbles' spatial re-distribution. Consequently, two modes self-regulation were captured: (1) stress self-regulation as a result of pressure rise and fall due to thermal expansion and creep/thermal cycling, respectively, and (2) temperature self-regulation due to the locally enhanced thermal conductivity, k, in the core region of the bed accompanied by locally deteriorated k in the surrounding region. These two mechanisms are desirable as they lower the probability of the events of pebbles crushing/bed creep and thermal runaways under high temperatures/stresses and poor heat extraction.

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P3.162 Tritium transport model at breeder unit level for HCLL breeding blanket

The Helium-Cooled Lithium Lead (HCLL) breeding blanket is one of the European blanket designs proposed for DEMO reactor. A tritium transport model is fundamental for the correct assessment of both design and safety, in order to guarantee tritium self-sufficiency and to characterise tritium

con-centrations, inventories and losses. The present 2D transport model takes into account a single breed-er unit located in the outboard equatorial module of the HCLL breeding blanket, which is one of the most loaded modules in normal operating conditions. A multi-physics approach has been adopted considering several physics phenomena, providing for buoyancy effect, temperature fields, tritium generation rate and velocity profile of lead-lithium and coolant. The transport has been modelled considering advection-diffusion of tritium into the lead-lithium eutectic alloy, transfer of tritium from the liquid interface towards the steel (adsorption/desorption), diffusion of tritium inside the steel, transfer of tritium from the steel towards the coolant (recombination/desorption), advection-diffusion of diatomic tritium into the coolant. Moreover, a preliminary evaluation of the magneto-hydro-dynamics effect (MHD) has been also performed. Tritium concentrations, inventories and losses have been derived under the above specified phenomena. Results, input and boundary conditions are illus-trated in detail within the paper.

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P3.172 A set-up for studying synergistic isotope effects in the permeation processes of hydrogen isotopes through metals

Tritium permeation into the structural materials and further in the coolant of the fusion devices is one of the most important safety issues. Various mathematical model and experiments have been carried out to estimate the amount of tritium permeated in the key components of the fusion devices. However, some issues related to the permeation of hydrogen isotopes through metals, like those regarding synergistic isotope effects (due to the competition and the coupling of the various isotopes), are still unresolved. Thus, to further explore such issues, a research project has been started by the Experimental Pilot Plant for Tritium and Deuterium Separation from ICSI Rm. Valcea. The aim of the project activities is to develop a simulation tool that eventually will be benchmarked against the experimental results. The co-existence of two or more isotope species in the permeation processes will be investigated basically considering the variation of the partial pressure of the hydrogen species on both sides of the permeating plates. In a further step the activities will be focused on the specific cases related to the needs for the EU DEMO Breeding Blanket and in particular targeting the tritium permeation from the He purge gas into the He coolant for the HCPB option. These activities can also be extended for the issues related to the tritium permeation from LiPb into the cooling water. In this paper we will discuss some modelling approaches that takes into account the isotope effects due to the competition and the coupling of the hydrogen isotopes that occur through surface processes, and that could have effect on predicted permeation rates, and also we will describe a versatile installation designed to experimentally investigate these effects, by various methods of testing.

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P3.165 Estimation of thermophysicochemical properties in uranium-hydrogen (U-H) system

Uranium (U), which has three allotropic crystal modifications (alpha-, beta-, and gamma-U), is a strong candidate medium for storing and delivering hydrogen or hydrogen isotopes. Alpha-, beta-, and gamma-U are stable at a temperature of up to 668℃, from 668℃ to 775℃, and above 775℃, respectively. Because the temperature of the uranium hydride (UHx) formation is limited at room temperature or below 250℃, it is assumed that all uranium is in the form of alpha-U. UHx has two allotropic crystal modifications (alpha- and beta-UHx). A common beta-UHx is produced as a single phase above 200℃. In this study, thermophysicochemical properties of the uranium-hydrogen (U-H) system were estimated by introducing an alpha-phase orthorhombic A20 crystal lattice structure in uranium (alpha-U) and a beta-phase cubic beta-tungsten A15 crystal lattice structure in uranium hydride (beta-UHx) and applying the chemical mixing rules. An assumption of hydrogen intervention

into the interstitial sites of the orthorhombic symmetry crystal lattice of alpha-U was used to obtain a plausible property estimation. Experimental data in rarity in U-H system were used to calculate and correlate the consistency of the thermal, physical, and chemical properties of the complex atomic structure in a unit cell. As a result, the volume expansion of the beta-UHx was greatly influenced by the hydrogen content, but showed no meaning in the thermal expansion within the engineering concept. In consideration of the heat capacity, the temperature effect from hydrogen - an interstitial heat quantity - in the beta-UHx formation was mainly the attributed factor, but not the hydrogen content.

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P3.167 Tritium separation via high-temperature proton-conducting ceramic pumps

This paper is aimed at addressing critical issues related to tritium separation in fusion reactors. One of the effective tritium separation technology is using high temperature proton conducting materials as hydrogen isotope separation membranes. When a direct current is applied to the electrochemical hydrogen pump, hydrogen and its isotopes in the anode side can be electrochemically extracted to the cathode side in spite of the chemical potential difference. A large one-end closed proton-conducting ceramic tube of CaZr0.9In0.1O3-α, which had the size of 19.6 mm in outer diameter, 17 mm in inner diameter and 520 mm in length, was used in this work. The porous Pt electrodes were attached on the inner (anode) and outer surface (cathode) of the tube by a pasting method, and its inner and outer length in axial direction was 520 mm and 400 mm from the closed end, respectively. When the supply gas containing 100 ppm and 0.85 V DC was applied to the hydrogen pump, almost all 100 ppm H2 in the anode gas has been entirely extracted, i.e., the hydrogen recovery efficiency was more than 99%, suggesting that the extraction of hydrogen could be operated with a current efficiency close to unity. The tritium extraction performance was also evaluated from a mixture of gases including 960.9 ppm hydrogen and radioactive tritium concentration of the order of 10 Bq/cm3 balanced with helium at 1023 K and applied 0.80V DC, which indicated that more than 31% of tritium supplied was recovered, and diffusivity of hydrogen in the proton conductor ceramic is larger than that of tritium.

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P3.170 Enhanced tritium production in fusion-fission hybrids for the external consumption

While the future fusion power reactors will consume and reproduce tritium for their operations, essential amounts of tritium will be required from external sources for their initial start-ups in the commissioning periods. Up-to-date evaluations of the start-up inventories are comparable with or even exceed the available commercial tritium resources in the world nuclear industry. At present the neutron rich fusion-fission hybrid systems with subcritical blankets are generally considered as possible producers of several products as nuclear fuel, energy and neutrons for waste burning. In this study, the hybrids were regarded as multifunctional systems including also a function of expanded tritium production. Some intrinsic properties and distinctive of such systems as nuclear performance variation in operation periods were remarked. It is shown that a surplus tritium of a few kilograms may be produced even in a relatively small of a DEMO scale fusion driven neutron source during 3-4 operation years that exceeds essentially the tritium inventory from a commercial CANDU type heavy water nuclear reactor. The assessments showed that a fusion – fission hybrid with changeable blankets and/or mobile blanket compositions, apart from the known nuclear-engineering feasibilities, can come up into place of an efficient tritium supplier in the developing fusion power industry.

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P3.174 New catalytic packing performance, theoretical and experimental characterization for LPCE process

The research activity for development of catalytic package that equips water-hydrogen catalytic isotopic exchange columns was of permanent interest for the Institute's research team, mainly motivated by the integration of the Liquid Phase Catalytic Exchange (LPCE) process in most of the detritiation technologies for tritiated water generated from nuclear reactors.
In recent years, our research team has been developing an order mixed catalytic package, structured type, which makes it possible to use it in high-load isotopic exchange installations due to low pressure drops, providing a continuous isotope exchange process.
The paper presents a structured catalytic package consisting of the B7 ordered hydrophilic packing made of stainless steel developed/patented by ICSI and a PT/C/PTFE hydrophobic catalyst, having an arrangement after a geometry chosen to achieve a high isotopic transfer performance.
In order to characterize the new mixed catalytic package, a laboratory installation equipped with this type of package was designed and built.
The research has been conducted to obtain fluid flow data through mixed catalytic package and also hydrogen isotopes separation performance.
Laboratory research in order to assess the isotopic exchange performance was carried out in two stages for the transfer from water to gas of deuterium and respectively tritium.
The paper presents the experimental installation, the experimental data obtained for the deuterium transfer and the performance parameters characterizing the mixed catalytic package.
Performance parameters are determined from the gas phase concentration profile determined from the concentration measurements at 5 sampling points along the catalytic isotopic exchange column.
These performance parameters are the reference data for the catalytic package that are used in the mathematical models which represent water-hydrogen catalytic isotopic exchange for simulation and for the isotopic exchange columns sizing. These data will be compared to the values obtained at tritium transfer.

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P3.175 A complete EM analysis of DEMO WCLL Breeding Blanket segments during VDE-up

Within the EUROFusion consortium, a big effort is made in order to analyze the electromagnetic loads that act on the in-vessel components during normal and off-normal operations, being an important input for their structural assessment. With regard to the Breeding Blanket (BB) project, a global DEMO EM model, feasible to account for different blankets design, has been developed last year with the capability to analyze EM transients in presence of both toroidal and poloidal magnetic field and considering materials with non-linear magnetic properties. Using the FE model based on the WCLL design, a VDE-up with a 74 ms current quench time was analysed. The main differences of the present with the previous analyses, only focused on eddy currents (and thus, on the related Lorentz's forces), are the evaluation of:
1) the effect of the electrical connections of the port plugs with the VV;
2) the interaction of the BB magnetized material and magnetic field (ferromagnetic forces).
The first analysis showed a significant change in the eddy currents path especially during the thermal quench, underlining the importance to consider carefully the electrical contacts with the VV. For what concern the second point, being the structural material of the BB segments, EUROFER, a ferromagnetic material with a magnetic saturation of ~1.8 T, relevant ferromagnetic forces (in the range between 5 and 10 MN in the radial direction) have been found to act on the Inboard and Outboard segment. This result becomes of particular importance especially for the definition of the attachment system with the VV. Furthermore, the contribution due to Halo currents has also been analyzed, giving thus a complete view of the EM loads behavior during the considered off-normal events.

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P3.176 MeV energy ions accelerators for simulation the neutrons damage in fusion reactor materials

Fusion reactors materials (FRM) will be exposed to 14 MeV fusion neutrons and damaged up to 15 dpa/year. The investigation of neutron irradiated materials is possible only in special conditions in a hot cell. The MeV-range energy ions can be used to simulate the effect of neutron-induced damages in FRM. Such simulation experiments can be used to study the effect of displacements on the materials structure and hydrogen isotopes retention properties.
In the work the experimental technique for MeV energy ions irradiation of FRM at NRC "Kurchatov Institute" – ITEP will be described and preliminary results of tungsten irradiation by Fe ions and protons will be presented. Two MeV-range linear ion accelerators uses for FRM irradiation: Heavy Ion Prototype (TIPr) with a 5.6 MeV Fe ions and I-2 with a 22 MeV protons. The accelerators were modified for simultaneous irradiation of 4 samples of 10×10 mm at controlled elevated temperature. The use of 5.6 MeV Fe+ allows to create displacement to damage level 1 dpa during 10 hour of the near-surface layer (1.5 μm for tungsten). A thin damaged layer is suitable for investigation of a material structure, but it is inconvenient to study the hydrogen retention in the damaged layer. The heavy ions like Fe+ creates cascades of displacements with an inhomogeneous depth distribution while neutrons produce uniformly distributed defects. The light ions like protons leads to produce damages at depth up to 400 μm for tungsten with a uniform defect distribution that makes it possible to study the hydrogen isotopes retention in defects by many methods. The disadvantage of using high energy protons for W irradiation is a generation 183Re and 184Re with a half-life of 70 and 38 days respectively.
This study was supported by the Ministry of Education and Science of the Russian Federation (ID RFMEFI61317X0084).

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P3.177 Test facility for the measurement of tritium permeation through fusion power plant materials

A DEMO fusion reactor will need to be self-sufficient in tritium fuel, with breeding planned within blankets surrounding the vessel. However, at the high temperatures within the breeding blanket region, tritium will readily permeate into the coolant through most materials of construction, causing a loss of this valuable commodity and contamination of the coolant stream. Special coatings have therefore been developed to minimise tritium loss.
A system has been designed and constructed at UKAEA to measure the efficacy of these coatings by direct measurement of tritium permeation at DEMO-relevant temperatures of around 800 K. This presentation explains the novel experimental design, driven by overcoming the challenges of handling a radioactive and explosive gas under such conditions.
High temperature components of the system are constructed from tungsten coated internally with beta-silicon carbide, due to its low tritium permeability at these temperatures. Detection is by a variety of methods to fully sample a range of sensitivities, and hence variations in material permeability. These include pressure rise into a known volume with residual gas analysis measurement, ionisation chambers, and beta-induced x-ray emission from a gold substrate on a beryllium window. A series of containment vessels allows any tritium permeated from the primary containment to be recovered.
The experiment allows an easily-mounted sample to be placed within its housing, with minimal intervention over the duration of a test, estimated to be up to several weeks for the most resistant samples. The system could easily be adapted to measure permeation of any hydrogen isotope through morphologically flat samples at temperatures between ambient and 800 K. Additional detection mechanisms, such as liquid scintillation counting, may be required at lower temperatures for low-permeation materials.

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P3.179 Evaporation and condensation model for a cooling system of the LiPb cold trap device

The eutectic liquid metal LiPb is considered as one of the tritium breeders of the first fusion power reactors. The flowing liquid metal dissolves alloying elements of the structural steels and thus causes their corrosion. The proposed type of the cold trap is a device providing extraction of corrosion products from liquid metal by gravity separation, which occurs at lower temperatures than the operating conditions of LiPb in the fusion reactor blankets.
The developed cold trap consists of three loops. The primary loop is for LiPb liquid metal flow purifying. The proposed cooling of the primary loop is based on the water-steam natural convection with two basic principles - evaporation and condensation. An evaporation section supports the optimal heat transfer from the primary loop, and, for the final heat removal from the trap body, the condensation section then transfer the accumulated thermal energy to the tertiary loop via a condenser. Due to high maintained temperatures of the liquid metal, the high saturation conditions with the corresponding temperatures in the secondary loop can be expected. To ensure reliable conditions at all operational thermal loads the presence of the inert gas (argon) in the secondary loop is provided. However, adding of the inert gas into the steam significantly reduces the heat and mass transfer to the condenser and consequently leads to degradation of the heat removal potential of the condensation. A model based on dimensionless numbers is developed and applied to solve these processes. Validation of this model for the operational conditions and achieved results for the cold trap operational states are analysed and discussed.

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P3.180 Neutronics Analysis of fusion breeding blanket concept to reduce peak of nuclear heat distribution

The high peak value of nuclear heat distribution in the fusion breeding blanket is expected to make cooling system design difficult for DEMO. The maximum peak value of about 10 W/cm3 is assumed in the Test Blanket Module with the maximum operational power of 700 MW in ITER. The peak value of nuclear heat distribution in the blanket of DEMO will be increased in proportion to the operational power and a few hundreds of W/cm3 are estimated as the maximum peak in the blanket of DEMO with the operational power of a few thousand of MW.
In this study, neutronics analysis for the blanket design reducing the peak value of radial nuclear heat distribution was performed considering the operation power of DEMO. The peak nuclear heat is generated in the breeder mainly due to the nuclear reaction of Li-6 for the blanket design of the separated breeder and multiplier composition. Therefore, the blanket concept mixing the functional materials of breeder and multiplier were considered to reduce the nuclear heat peak in the breeder. At first the Liberite was considered as the functional material of the blanket. The Liberite, Li2BeSiO4, is a mineral including Li and Be. Secondly, a functional material similar with TRISOL which is one of the fuel types used in the VHTR, was considered. It is assumed that the Li pebble (Li2TiO3) with Be coating is used as the functional material of the blanket. The tritium production or tritium breeding and nuclear heat distribution were evaluated with the proposed blanket design. Additionally, re-loading of the spent functional material was evaluated based on the Li burn-up calculation, considering long-term operation and production of radioactive waste.

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P3.181 Numerical analysis of heat transfer characteristics for pebbles bed with variable fluid flow and packing factor conditions

The contact resistance effect in the interface between pebble beds was studied with CFD analysis. The lithium ceramics is used as breeder with the form of sphere-shaped pebbles for the extraction of the tritium in some Test Blanket Module (TBM) candidates of ITER. The flow of the gas is essential for the extraction of the tritium. The effects of the gas flow was considered. The effect of fluid flow was not significant on the maximum temperature of the pebble bed. The fluid is the gas phase and the removed heat by the fluid flow is minor. The packing factor was determined by the arrangement of the pebbles in the layers. The packing factor was adjusted by removing the pebbles to make the void in the pebbles bed. Although total generated heat is reduced as the total volume of the pebble becomes smaller, the maximum temperature of the pebbles bed was increased. The thermal conductivity of the pebbles bed was estimated for all results and compared with the some correlations. As the results of comparison, the overall tendency agrees for the effects of the packing factor parameter.

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P3.183 Design prototyping and manufacturing of Johnston Couplings for the Cryopump for the MITICA test facility

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The ITER project will require large cryopumps of flat-geometry to pump the Heating Neutral Beam Injectors (NBI), and similar cryopumps to pump the diagnostic Neutral Beam (DNB). The cryogenic supply uses supercritical Helium for the cryopanels and gaseous Helium for the thermal shields of the cryopumps. The cryogenic fluids will be produced by a large cryogenic plant, and then distributed by means of cryolines from the Cold Valves Boxes (CVB) to the cryopumps.
The first ITER Neutral Beam ITER cryopump will be built for the full-scale prototype of the ITER heating neutral beam injector (MITICA experiment at PRIMA-NBTF), under realization in Padua, Italy.
In order to facilitate repair or removal of a cryopump, cryogenic removable connections (named Johnston Couplings) in an ITER specific design are conceived as interfaces between the cryopumps and the cryogenic distribution system of the MITICA test facility. These Johnston couplings when used on ITER will have a nuclear confinement function and hence require to be built to allow fully volumetrically examinable welds on the confinement boundary. This coupled with the need to achieve low cryogenic heat loads and be able to be installed in different orientations has led to a highly innovative design.
This paper deals with the development of Johnston Couplings (8 pairs) to be used for the very first ITER NBI cryopumps of the MITICA test facility. The design, validation, manufacturing, and factory acceptance testing activities are described, as well as the issues faced and technical solutions adopted to fulfil the ITER requirements.

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P3.188 Numerical influence analysis of the packing structure for ceramic Li4SiO4 and Li2TiO3 pebble beds

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As an integral part of the solid blanket, ceramic tritium breeder pebble bed plays a vital role in tritium breeding during the operation of the fusion reactor. The packing structure of the pebble bed has an impact on its thermomechanical behavior and tritium exaction in the solid blanket, which is actually affected by packing methods, particle size, container size, particle smoothness, particle hardness and so on. Considering the application of pebble bed in helium cooled solid blanket (HCSB) of Chinese Fusion Engineering Test Reactor (CFETR), a randomly packed pebble bed with a 1mm diameter particle is adopted in the HCSB tritium breeder module. The effect of the container size, the friction coefficient and restitution coefficient between the particles on the packing structure of Li4SiO4 and Li2TiO3 pebble beds need to be analyzed for the HCSB. This paper uses Discrete Element Method (DEM) for numerical analysis, and the simulation model similar to the shape of tritium breeding pebble bed in the solid blanket is generated by pouring the particles into the cuboid container under the gravity. The effects of L/d ratios (L means the length of cuboid pebble bed and d refers to the diameter of the particle), friction coefficient and restitution coefficient on packing structure of the mono-sized Li4SiO4 and Li2TiO3 pebble beds are studied. The curves that the value of packing factor of pebble bed varies with those factors were obtained. Besides, the distribution of the local packing factors in the pebble bed and the layered arrangement of particles near the wall were analyzed, which can provide a reference for the design of the solid blanket.

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P3.189 High temperature hydrogen selective solid-state electrolyte sensor fabricated by slip casting

Tritium management is one of the main challenges that future nuclear fusion energy has to achieve. Accurate tritium monitoring is a basic task in order to have relying fusion reactors. High temperature sensors have to be developed to make this monitoring a reality. Hydrogen sensors based on solid-state electrolytes can be a reliable option to perform this monitoring. These types of sensors offer resistance to harsh chemical environments, temperature depending conductivity and quick and easy to measure signals.
Potentiometric and amperometric hydrogen sensors based on solid-state electrolytes were previously studied at the Electrochemical Methods Laboratory at Institut Químic de Sarrià (IQS) at Barcelona. These previous sensors contained a pellet-shaped solid-state electrolyte. Performing a slip casting process to the electrolyte can increase the sensor signal due to a surface increase. The satisfactory results obtained with shaped amperometric sensors by slip casting present a promising path to the development of high temperature tritium sensors.
In the present work, a process to fabricate solid-state electrolytes by slip casting for amperometric hydrogen sensors has been developed. Amperometric electrochemical measurements have been performed at different hydrogen partial pressures (0.25 to 2.5 mbar), at 500 °C and applying 5 V between the electrodes of the sensor to polarize it.

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P3.193 Study on the anisotropy in microstructure and tensile properties of the 12Cr ODS steel

Oxide-dispersion-strengthened (ODS) steels have been developed as one of prospective candidate materials for fast reactor cladding as well as fusion reactor blanket applications. The anisotropy in microstructure and tensile properties in the range from room temperature (RT) to 973 K of the 12Cr ODS steel with the nominal composition of Fe-12Cr-2W-0.3Ti-0.25Y2O3 (in wt.%) was investigated by scanning electron microscopy, electron backscatter diffraction technique, transmission electron microscopy, Vickers hardness and tensile tests. The hardness was measured at RT with a load of 1 kg and a dwell time of 15 s. SSJ specimens for tensile tests with gauge section of 5 mm (L)×1.2 mm (W)×0.25 mm (T) were machined along rolling direction (RD) and transverse direction (TD). The tensile properties were examined with an initial strain rate of 6.7×10−4/s. Results show that mi-

crostructure was not homogeneous, and consisted of coarse recrystallized grains and fine elongated grains with high density of dislocations, which gave rise to the deviation of Vickers hardness. Texture was also clearly characterized, which was <110> axis parallel to the RD, mainly {001}<110> and {111}<110> components. The tensile properties in the RD were better than that in the TD, especially for the elongation. The fracture morphology at RT in RD and TD revealed a mixture of flat cleavage planes and shallow dimples, and a majority of elongated cleavage planes, respectively. The ultimate tensile strength and yield strength were decreased as the testing temperature increased. The fracture morphology and the damage mechanisms in the RD and TD will be discussed in detail.

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P3.194 Dielectric characterization of up to 0.1 dpa neutron irradiated insulating materials for fusion applications

Functional materials have diverse applications in fusion reactors and it is clear that insulators are among the most versatile groups. They are the base of all the electric and radiofrequency systems in diagnostics and heating systems from DC to very high frequencies (RF, H&CD, NBI…). Additionally, insulators are subjected to quite different conditions (voltage, temperature, frequency…) depending on their application. In order to be a good candidate for dielectric applications, a material needs to be an excellent insulator at the required frequency, temperature and voltage. Furthermore, it has to exhibit exceptional radiation resistant behaviour because radiation levels can reach important levels for some in-vessel applications. Finally, reproducibility at industrial scale must be achieved for those materials to operate at a reactor level. In the European WPMAT-FM subproject, a set of samples acquired from different high-performance manufacturers have been subjected during 2017 and 2018 to various neutron irradiations to test their dielectric properties after neutron damage at higher doses than available in literature, to fill this gap.
We present pre-irradiation and PIE results coming from a neutron irradiation with 2 sets of dose and temperature specifications: an irradiation with a 0.9-1.0x1020n/cm2 dose at 260-290ºC and an irradiation with a 0.7x1020n/cm2 dose at 160-190ºC. These doses provide relevant data for both ITER and DEMO design parameters for many applications. Measurements include loss tangent and permittivity. Loss tangent is a key parameter to evaluate the power loss in the insulator (and hence its heating)

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P3.195 Properties of low friction anti-seize coatings for fusion applications

Remote maintenance in fusion machines such as JET and ITER relies on sliding interfaces such as bolted joints. Experience in JET, where removal torques much higher than installation values with uncoated bolts is commonplace, led to the installation of experimental bolted assemblies in 2015: the first of its kind in JET. These assemblies included some 660B stainless steel ITER Blanket-specific bolts with a solid sputtered coating of MoS2. After removal in 2017, tests revealed no pre-load loss and undoing torques similar to the original installation torques: thus confirming that the low friction and anti-seize properties were indeed realised in a tokamak environment.

Parallel activities include measuring the fundamental properties of such coatings, including outgassing, friction and sensitivity to humidity.

A UHV (Ultra High Vacuum) facility (<3 x 10-10 mBar after baking to 200 ⍰C), engineered in the MRF (Materials Research Facility) with a vacuum load-lock facilitates highly sensitive outgassing measurements. Tests have revealed small, but measurable, quantities of H2S from coated samples.

Sulphur containing gases are a serious plasma pollutant, so quantifying the outgassing is imperative.

Such coatings are normally used in a dry state but are believed to be moisture-sensitive and could be accidently exposed to moisture in a tokamak. Hence, humidity-exposed samples are also being tested. The friction properties of the samples, along with unexposed control samples, will be measured using the PoD (Pin on Disc) method.

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P3.197 FNG Copper Benchmark Evaluation for the SINBAD Database

Shielding Integral Benchmark Archive and Database (SINBAD) project started in the early 1990's at the Organization for Economic Cooperation and Development's Nuclear Energy Agency Data Bank (OECD/NEADB) and the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory (ORNL) with the goal to preserve and make available the information on the performed radiation shielding benchmarks. SINBAD is available from the NEA Data Bank and RSICC. The database comprises now 102 shielding benchmarks, divided into three categories, covering both low and inter-mediate energy particles applications: fission reactor shielding (48 benchmarks), fusion blanket neutronics (31), and accelerator shielding (23) benchmarks.
FNG-Copper benchmark and HCLL mock-up are the first evaluations being prepared after a long pause. The work has been partly funded by the Fusion for Energy - F4E under the Specific Grant Agreement F4E-395-01 and is planned to be followed by other evaluations. The neutronics benchmark experiment on a pure Copper assembly was also performed in the frame of the F4E Task at the 14-MeV Frascati neutron generator (FNG) of ENEA Frascati with the objective to provide validation of the copper nuclear cross-section data relevant for ITER design calculations, including the related uncertainties. A detailed evaluation of the experimental configuration, measurement system and results was performed in the scope of the SINBAD evaluation with a particular focus on a realistic, complete and consistent estimation of uncertainties involved in the measurements and the calculations. Several input models for Monte Carlo (MCNP5) and deterministic (DORT) transport and sensitivity/uncertainty (SUSD3D) codes are provided using both very detailed and simplified geometry models and different neutron source modelling. To facilitate the use of the experimental data for nuclear data validation and improvement, the SINBAD compilation includes also the sensitivity energy dependent profiles of the detector reaction rates with respect to the underlying cross-section data.

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P3.199 Development of W-monoblock divertor components with included thermal barrier interfaces

In the case of DEMO fusion reactor, the divertor should be able to extract a steady heat flux of about 10 MW/m2. A promising concept is the W-monoblock. which should be connected to a CuCrZr or an advanced Cu ODS alloy pipe passing through the W component. Taking into account the optimum operating temperature windows for W and existing Cu-based alloys and the thermal expansion coefficients mismatch of these two materials, a "thermal barrier" interface material is inserted in between in order to mitigate the thermal stresses and to optimize the heat flow through divertor components. In this work we investigate the feasibility to realize such divertor components using materials produced by FAST (field assisted sintering technology). This powder metallurgy technique was used firstly to produce W or W-based composites and the thermal barriers in an almost final shape and then to join the materials in realistic divertor mock-ups. The thermal barrier materials are various Cu-based composites [1,2] which are included both as single material or as

functionally graded components. The interface quality between different materials is investigated by scanning electron microscopy and the heat flow through components is evaluated using simulations.
[1] M. Galatanu, M. Enculescu, G. Ruiu, B. Popescu, A. Galatanu, Fus. Eng. Des. 124 (2017) 1131.
[2] M. Galatanu, M. Enculescu, A. Galatanu, Fus. Eng. Des. 127 (2018) 179.

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P3.20 2Development of a novel irradiation chamber to assess the influence of magnetic fields on radiation damage in materials

To date the research on structural materials for future fusion reactors has been focused on the evolution of mechanical properties with irradiation dose, energy, temperature, etc. However, the performance of materials irradiated under the presence of magnetic fields remains unclear. This aspect becomes critical, as structural materials in fusion reactors will need to withstand intense and hazardous radiation environments under the presence of strong magnetic fields. In principle, material micro-structural and mechanical properties are modified by radiation-induce propagating defects. It is hypothesized that such propagation is sensitive to magnetic fields. Therefore, experiments are essential to investigate mobility, recombination, clustering or dissociation of defects, hydrogen and helium creation in structural materials. Indeed, recent irradiation experiments point to external magnetic fields as a key parameter in Cr mobility. Experiments to clarify this have been already carried out under external field strengths of around 0.4 T and temperatures up to 100 °C, however this values are well below those expected in future reactors.
In order to reproduce fusion relevant conditions, a sample irradiation chamber consisting of a magnetic flux concentrator (around 1 T) and a sample holder with an oven mounted on a manipulator, have been developed for the Danfysik implanter at Ciemat. In order to minimize magnetic flux losses a magnet is close coupled to both, the sample and oven in this compact system. A magnet temperature below 150 °C, when the sample is at 450 °C, is guaranteed by means of a thermal shield (thermocouples monitor both temperatures). The magnetic flux concentrator and manipulator have been fabricated and initial tests are underway. The design, along with first results from irradiation studies of structural material alloys (FeCr) will be presented. For this, samples will be helium implanted with/without magnetic field at high temperature. Implanted samples will be studied using several techniques.

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P3.204 Early results on deposition of W-Cr alloy by industrially available large scale technologies

Tungsten has many advantageous features; however, it is rather susceptible to oxidation at temperatures above 500 ℃. By the addition of various oxide-forming elements to tungsten, self-passivation is induced. During exposure of the alloy to air, a passivation layer is formed on its surface, thereby preventing further tungsten oxidation, material degradation and related radiation spreading. Currently, the most promising result has been achieved using W-Cr solid solution with a small addition of yttrium prepared by mechanical alloying and Field Assisted Sintering.
The aim of the present study is to explore the feasibility of two large-scale technologies that are either already available in the industry or technologies where the laboratory experiences can be easily translated to industrial production. Both methods can fabricate the plasma-facing layer directly on the structural part, without the need for joining. The following technologies were applied to deposit W-Cr coatings:
Cold Spray is a coating deposition method that offers a huge versatility and material throughput without the necessity for any special substrate surface pre-treatment. Using kinetic energy in lieu

of thermal input, the method is particularly good for deposition of metals or metallic blends. Low temperatures involved in this process prevent material oxidation.
Radiofrequency plasma spraying is a coating deposition method that operates in a controlled atmosphere (inert gas, reactive gas or vacuum), effectively eliminating oxidation of the molten metal. The temperatures involved in this process are high enough to melt tungsten and ensure its proper deposition
The first results clarify the capabilities and limits of these technologies with respect to production of W-Cr solid solution deposits. Basic questions were answered, e.g. whether a specific powder treatment is necessary prior to the deposition, what is the most suitable powder size for production of the solid solution or whether a post-treatment of the deposits is necessary.

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P3.206 Characterization of ZAO® sintered getter material for use in fusion applications

The use of non-evaporable getter (NEG) pumps is common in many UHV applications including surface science, analytical instruments and very large vacuum systems for high energy physics. In the past years, getter solutions based on the new sintered alloy ZAO® have been developed enabling operation in the HV regime, i.e. 10-6 Pa and above. The properties of this NEG material make it appealing for applications dealing with large fluxes of hydrogen and its isotopes, as typically occurs in fusion research. In particular, the use of NEG pumps is interesting for the vacuum system of negative ion-based neutral beam injectors, commonly used in fusion devices for plasma heating, which require a huge deuterium gas throughput to sustain a dynamic equilibrium between low-pressure beam drift regions and a relatively high-pressure neutralizer cell. The key advantages of a NEG pumping solution for hydrogenic species are the high specific pumping speed and capacity, the robustness, intended as ability to withstand a large number of adsorption/desorption cycles with unchanged pumping properties, the ease of integration. In addition, NEG do not release hydrogen unless power is supplied to heat them; this property allows to control the rate of hydrogen release during regeneration, i.e. H2/D2 extraction, but also addresses an important safety issue related to uncontrolled release in case of power outage or subsystem failures.
In this paper we report the experimental characterization of ZAO® sintered getters, in pressure regimes and sorption amounts relevant for the use in the next generation NBI. Discs and pumps were characterized in different conditions of pressure between 10-4 and 10-1 Pa and operating temperatures up to 150℃, with respect to the equilibrium isotherms, adsorption and desorption of H2/D2 and cyclic embrittlement limit. In particular, the feasibility of getter regeneration within time windows acceptable for a high-availability NBI system was demonstrated.

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P3.208 Creating high-resolution simulations of manufactured divertor components from X-ray and neutron tomography

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The ability to estimate the in-service performance and lifespan of components is key to realising a commercially viable fusion energy device. The finite element method (FEM) is used to estimate performance of a component design with computational simulations. Image-based FEM (IBFEM) converts 3D images (e.g. X-ray tomography) into high-resolution models for part-specific simulations

that account for minor deviations from design due to manufacturing processes (e.g. machining tolerances or micro-defects). To investigate the potential of this technique for fusion applications, IBFEM was used on three samples: (1) ITER reference divertor monoblock; (2) CCFE 'thermal break' DEMO monoblock concept; (3) IPP tungsten fibre / copper matrix composite. 3D imaging was performed with high-power X-ray tomography and neutron tomography for a comparison. 3D images were converted into high-resolution IBFEM meshes using the software ScanIP. Thermo-mechanical analysis was performed with ParaFEM using boundary conditions that simulated reference in-service divertor conditions. IBFEM simulation results successfully resolved the impact of minor deviations on performance. For example, in the CCFE monoblock, void regions existed in the braze layer which behaved as micro thermal barriers at the material interface between the coolant pipe and tungsten armour. In the IPP composite pipe, the exact positioning of the tungsten fibres caused variations in the location and value of peak temperatures on the armour's plasma facing surface. This work demonstrates the value of the technique's capability to digitally test 'real' components. Because results can be interrogated through the sample's full volume, this has the potential to yield more sophisticated information than pass/fail experimental tests. This could be used to individually rate each component, in a manner similar to material purity, so that the best performing parts could be placed in the most demanding regions of the tokamak.

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P3.210 Using sodium as filling and heat-conducting material in the irradiation capsules of the IFMIF-DONES

The capsules of the IFMIF-DONES High Flux Test Module (HFTM) are packed densely with Eurofer specimens. A filling material (previously NaK-78 and presently sodium) is needed to fill any empty volume to improve the heat conduction and obtain uniform temperature distribution. Sodium is replacing NaK-78 because potassium generates argon isotopes leading to a pressure increase and formation of bubbles which counteract the purpose of introducing NaK-78 as a heat conductor. In this experimental study, two setups are used to investigate sodium as the new filling material. The first setup is dedicated to test the wettability of various specimens of Eurofer and stainless steel 316L by liquid sodium within the temperature range of 100˚C to 430˚C. The second setup consists of: (i) a full scale prototype of the HFTM capsule packed with Eurofer specimens, (ii) a stainless steel container for melting the sodium, and (iii) heaters, thermocouples and supporting parts. The objective is to demonstrate successful sodium filling and emptying of the prototype capsule and to determine how well the specimens are wetted and the empty volumes are filled with sodium. All experiments are performed inside a glovebox filled with dry argon to maintain an oxygen- and moisture-free atmosphere. The results of these experiments, which are relevant to IFMIF-DONES HFTM capsules and other sodium-using applications, are presented and discussed in this paper. For example, the results showed that efficient wettability of Eurofer and stainless steel 316L specimens by sodium may be assured when the temperature of the steel-sodium system is 430℃ or higher.

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P3.212 Microstructure of oxide dispersion strengthened copper fabricated by mechanical alloying and hot isostatic pressing

A copper (Cu) alloy, having a high thermal conductivity, is a promised material for heat sink of diverters in a force free helical reactor (FFHR). Recently, Hishinuma's group succeeded in fabrication of oxide dispersion strengthened (ODS) Cu alloys using mechanical alloying (MA) and hot isostatic pressing (HIP) process. ODS is expected to bring about high-temperature strength and irradiation resistance to Cu alloys. The three important properties for their practical use thermal conduction, high-temperature strength and irradiation resistance, which have strong dependence on the internal structure. We therefore investigated the internal structure of ODS-Cu, fabricated by MA-HIP process.

The precursor powders of ODS-Cu were prepared by ball-milling the initial powders at 250 rpm for 32 hrs in an argon (Ar) atmosphere. We then mixed Cu, yttrium (Y) and CuO powders with a mass ratio of 98.17 : 0.79 : 1.04. Here, the samples is labeled Cu-1wt%Y according to the mass ratio of Cu and Y2O3. The mixed precursor powder was sintered by a HIP process at 950˚C for 1 hr under the pressure of 150 MPa. The internal structure was observed using electron microscopy.
Macroscopic observation showed that Cu-1wt%Y bulk was consisted of large Cu grains of more than 10 μm surrounded by fine Cu grains smaller than 1 μm. Many of Y2O3 particles of about 20 nm were distributed on grain boundaries of fine Cu grains, which presumably to act to inhibit grain boundary movement during heat treatment. On the other hand, our microscopic observation have shown that fine Y2O3 particles of 5 nm were dispersed in the fine Cu grains. These observation suggested that ODS-Cu alloys fabricated by MA-HIP process possess inhomogeneous structure, which is necessary for enhancing the aforementioned three important properties for applications as a diverter material.

P3 / 1261

P3.214 Thermophysical and mechanical properties of W-Cu laminates produced by FAST joining

W-laminates are multi layered composites realized from alternately stacked W and a second metal foils. Such materials are promising candidates for W-based structural materials for fusion reactors like DEMO or beyond concepts, due to the fact that cold-rolled ultrafine-grained thin W foils show exceptional properties in terms of ductility, toughness and ductile to brittle transition (DBT), in contrast to classic bulk W materials [1]. Therefore, different routes to transfer the W foils properties to bulk materials have been investigated [2,3]. In this work we present the results obtained for W-Cu laminates produced via a FAST (Field Assisted Sintering Technique) joining route. The main advantages of FAST resides in the short processing time, with subsequent lower recrystallization detrimental effects. Structural, thermophysical and mechanical properties of W-Cu laminate samples produced in this way are investigated and compared with those available for diffusion bonded similar samples.
[1] J. Reiser, L. Garrison, H. Greuner et al., Int. J. of Refractory Met. & Hard Materials 69 (2017) 66.
[2] J. Reiser, M. Rieth, B. Dafferner et al., J. Nucl. Mat. 423 (2012) 1.
[3] J. Reiser, M. Rieth, A. Moeslang et al., J. Nucl. Mat. 436 (2013) 47.

P3 / 1262

P3.215 Er2O3 coatings of different structures and their corrosion resistance of type 316 stainless steel to liquid lithium

Er2O3 coatings with different structures were deposited on type 316 stainless steel substrates by magnetron sputtering and corroded by liquid lithium for corrosion resistance study. The microstructure of the Er2O3 coatings was controlled by using two different methods, one the Er metal layer was deposited and oxidized successively, and the other directly by sputtering with Er2O3 deposition. Laser-induced breakdown spectroscopy (LIBS) technique was used to study the corrosion depth profiles represented by the longitudinal element distributions especially for that of lithium in the corroded layer of the specimens. It is observed that the Er2O3 coatings with Er metal layer being deposited and oxidized successively show an excellent corrosion resistance to the liquid lithium when corroded at 500 ℃for 250 hrs, but those prepared by directly sputtering with Er2O3 deposition show a poor corrosion resistance to the liquid lithium at the same corrosion conditions. XRD and SEM/EDX analysis for the corroded layers reveals the phase structure and morphology of the vertical sections, which gives results consistent with those obtained by LIBS analysis. Relevant model was proposed to explain the intrinsic mechanism of the difference in corrosion resistance to liquid lithium.

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P3.137 Initial Configuration Studies of the Upper Vertical Port of the European DEMO

In the current pre-concept phase of the European DEMO, integration studies of the systems in the Upper Port area are being carried out. In DEMO, the Upper Port of the Vacuum Vessel is extraordinarily large to allow for the vertical extraction of the Breeding Blanket segments. This requires a number of components inside and outside the port to be integrated with tight space constraints: The Upper Port structure and its annexes, the adjacent Toroidal and Poloidal Field Coils, the Thermal Shields, the piping connection to the Vacuum Vessel Pressure Suppression System, the Shield Plug and its inserts, the feeding pipework of the in-vessel components and part of the Breeding Blanket supporting structures. Apart from functional aspects, the design of these components is mainly driven by considerations of structural integrity, maintainability and irradiation shielding, which are mutually competing in many areas. A number of studies have recently been conducted on the design of the Upper Port and the required configuration of the components within. The present article describes the studied options and the respective results, the identified issues as well as the proposed engineering solutions, in particular with respect to the mechanical design of the Upper Port and the integrated Shield Plug.

P3 / 1267

P3.220 Modifications to the MELCOR-TMAP code to simultaneously treat multiple fusion coolants

This paper describes recent progress at the Idaho National Laboratory (INL) in developing the MELCOR-TMAP computer code for fusion. The MELCOR-TMAP for fusion computer code is being developed by the INL's Fusion Safety Program (FSP) [1] by modifying the US Nuclear Regulatory Commission's (NRC's) MELCOR [2] computer code for fission reactor severe accident analyses. The MELCOR code was chosen for application to fusion accidents because it is a well-established validated computer code that possesses the basic capabilities of predicting thermal-hydraulic transients while self-consistently accounting for aerosol transport in nuclear facilities and reactor cooling systems. Recently the INL FSP completed the process of merging INL's Tritium Migration Analysis Program (TMAP) with MELCOR to provide the US fusion community with a more comprehensive tool for analyzing accidents in future fusion reactors. However, prior to the present modifications, a MELCOR-TMAP user could only substitute one of a number of available fusion coolants for MELCOR's default coolant of water. The present version allows two additional coolants, plus water, to be included in a given accident analysis, provided that these coolants reside in different cooling loops or systems. This new capability corrects a modeling gap that is needed for fusion reactor safety assessments that contain more than a single coolant. In this article, we discuss the present code modifications, benchmarks MELCOR-TMAP against predictions from previous versions of the MELCOR code, illustrates the application of the code to analyzing an accident in a multiple fluids reactor concept, and describes plans for future modification of the MELCOR-TMAP code.

Note:
1This work was prepared for the U. S. Department of Energy, Office of Fusion Energy Sciences, under the DOE Idaho Field Office contract number DE-AC07-05ID14517.
2 Sandia National Laboratory in New Mexico (SNL-NM) is developing the MELCOR code for the US NRC.

P3 / 1268

P3.221 Design and analysis of the improved configuration of the Power Conversion System for the EU-DEMO power plant

DEMO is planned to be a prototype fusion power plant capable of demonstrating production of electricity at the level of a few hundred MW. DEMO is considered to be an intermediate step between the ITER experimental reactor and a commercial power plant. Design and assessment studies on the European (EU) DEMO are carried out by the EUROfusion consortium. The Primary Heat Transfer System (PHTS) transfers heat from the breeding blanket (BB) and other reactor heat sources (divertor and vacuum vessel) to the secondary circuit. Two BB concepts, and the respective PHTSs, for EU-DEMO are considered: the Helium Cooled Pebble Bed (HCPB) BB and the Water Cooled Lithium Lead (WCLL) BB. Two variants for each concept are possible: with or without the Intermediate Heat Transfer System (IHTS), including the Energy Storage System (ESS), between the PHTS and the Power Conversion System (PCS), which generates electrical energy. The role of IHTS and ESS is to prevent energy pulses removed by BB PHTS during the plasma burns to be directly transferred to PCS, to provide thermal energy to PCS more smoothly and continuously.
In 2017 we proposed the first concept of the PCS configuration for the option WCLL BB with the ESS, which ensured almost constant production of electricity during both plasma pulse and dwell phases. However, we observed some disadvantages of this concept, e.g., excessive temperature differences between the pulse and dwell phases in several heat exchangers. In the present paper, we propose the improved PCS configuration for the WCLL BB with the ESS option that still maintains almost constant production, and we analyze its operation during both the plasma pulse and the dwell phase using the GateCycle software, to show the system performance as well as the suitability of the concept.

P3 / 1275

P3.228 Studies on helium-3 separation at Cernavoda nuclear power plant

Helium-3 is a rare isotope of helium (1.37 ppm as fraction of total helium – natural abundance), with applications in medicine, industry, security, and science. Due to its high request, the world is experiencing nowadays a shortage of helium-3.
The most common source of helium-3 is the disintegration of tritium. Tritium is an unstable isotope of hydrogen, with a half-life of 12.3 years, and is a by-product in CANDU type nuclear reactors.
Tritium is found mostly in moderator heavy water. It is produced by the thermal neutron activation of deuterium atoms from the heavy water molecules. It can be found also in the cover gas used to control the pressure in the calandria and to provide a non-corrosive, non-radioactive atmosphere in parts of the system not filled with water.
In the first part, the paper propounds an analysis regarding the feasibility of helium-3 extraction from the cover gas used in Cernavoda Nuclear Power Plant. A theoretical calculus of the amount of helium-3 produced so far is presented. The production rate of helium-3 is related to the concentration of tritium in the moderator heavy water. The solubility of helium-3 in water is very low, therefore almost all of it will be found in the moderator cover gas.
The second part of the paper presents a technological solution for separation of helium-3 from helium gas at cryogenic temperatures.

P3 / 1271

P3.224 Project & Quality Management activities of ENEA's Fusion Program and use of the Primavera P6 software

ENEA through the Department of Fusion and Technologies for Nuclear Safety (FSN) actively par-

ticipates, playing a fundamental role, in the realization of ITER, contributing with the industry to the design and construction of many components ranging from diagnostic, power supply systems, superconducting magnets and Test Blanket Module auxiliary systems. The French Nuclear Safety Authority (ASN), has requested the Domestic Agencies (DA) of the seven Parties and, consequently, all the suppliers, to adopt a Quality Assurance Program. EU DA F4E defined the Supplier Quality Requirements in the QA document F4E-QA-115. ENEA fully meets these requirements, having a Quality Management System (QMS) in compliance with the UNI EN ISO 9001 standard since 2011. This paper describes the experience acquired at ENEA in the implementation of the QMS and the adoption of the Project Management (PM) approach in the execution of supplies for the Fusion program, applying quality and PM techniques to the phases of the project starting from the presentation of the proposal to F4E calls, the execution and the closing of the contract. The QM requirements, as for F4E-115, related to the organization of the activities, roles definition and responsibilities, risk management, document control, configuration management are fully in line with the provisions of ENEA's QMS. The required Quality Plans describe how ENEA implements the QMS during the progress of the works. These processes are documented providing objective evidence of their effectiveness. The PM approach is adopted to monitor the progress of the activities, to ensure that the contract's requirements are met and to maintain the evidence of the compliance. The tool adopted, according with the ITER's high level choice, is Primavera P6, the standard recognized on the market as a solution for the management of highly sophisticated and large-scale projects. This paper also describes the experience gained in its application.

P3 / 1279

P3.232 Radiation shielding requirements for the full power operation of the Linear IFMIF Prototype Accelerator (LIPAc) at Rokkasho

The IFMIF (International Fusion Materials Irradiation Facility) project aiming at material tests for a future fusion power plant is now in the 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 in order to demonstrate the validity of the low energy section of an IFMIF deuteron accelerator up to 9 MeV with a beam current of 125 mA in CW. The present study is devoted to finding the configuration of the radiation shielding in the accelerator building necessary for the operation of LIPAc at the full power of 1.125 MW to satisfy the radiation dose limit prescribed by the law. The beam accelerated in LIPAc is stopped at the beam dump, and a significant amount of neutrons and photons are generated. The detailed design and various studies for the beam dump have been conducted by CIEMAT, Spain, and most of its constitutional elements have been already manufactured in Europe. QST is then responsible for the installation on site and also for the nuclear licensing required for the operation of the accelerator. The accelerator building has the characteristic that it has many penetrations for HVAC ducts, cooling water pipes, cables and coaxial waveguides to supply the RF power to the accelerator. For this reason, some additional shielding shall be carefully designed to reduce the neutron streaming effect and to satisfy the dose criteria during full power operation with consideration for its construction possibility. In the present study, the detailed shielding analysis has been performed by taking the final configuration of the building integrated with the beam dump into account and the necessary condition for the radiation shielding has been found out.

P3 / 1280

P3.233 Generation of tritiated methane by reaction between tritium and methane on catalyst

Detritiation system (DS) is the key system to ensure safety of a fusion reactor. The DS must be designed to make sure of detritiation when an extraordinary event such as fire happens. Assuming that an accidental release of tritium and production of hydrocarbons by combustion of cables in case of fire occurs simultaneously, tritiated methane will be generated by the reaction between tritium and methane in a catalytic reactor of DS as a side reaction. If a large amount of tritiated methane was generated, a catalytic reactor kept the temperature at 773K for combustion of the tritiated methane was needed. However, a catalytic reactor at 773K has a risk to induce tritium permeation through the walls of the catalytic reactor and fire caused by temperature runaway of the catalytic reactor by heat of reaction of hydrocarbons combustion. In the present study, we investigated experimentally how to control generation of tritiated methane in a catalytic reactor for avoiding the risks. Our experimental investigation indicated that a generation rate of tritiated methane was sufficiently small, so that DS for extraordinary situation did not need a catalytic reactor at 773K for combustion of the tritiated methane. Details will be reported at the conference and the paper.

P3 / 1281

P3.234 Evaluation of the effect of impurities in concrete on shutdown dose and activation for fusion reactor facility

The exposure by shutdown dose and production of radioactive waste predicted from the activation analysis are interesting issues of fusion reactor facility design in the view of radiation safety. Impurities of the irradiated material, such as cobalt in the structural material, are occasionally an important factor in the evaluation of the induced activity.
Concrete is used as the neutron shield and structural material of the neutron production facilities like the nuclear power plant, accelerator facility, etc. The fusion reactor facility will also use the concrete as the shield and structure of the building. ITER uses normal concrete and heavy concrete as a structure material of the building and the bio-shield, respectively. It is also expected to be same in DEMO. Dominant radioactive nuclides produced from the normal composition of concrete such as Fe-55, Fe-59, Mn-54, Na-22, Na-24, etc. are considered not significant in terms of the contribution to the shutdown dose, because they are mainly beta decay nuclides.
In this study, the effect of impurities in concrete on the activation was evaluated based on the calculation of shutdown dose and induced activity. As the impurities in concrete, europium, cobalt, cesium, and nickel are considered. The shutdown dose and induced activities were calculated with the bio-shield composed of heavy concrete, in which highest neutron irradiation is expected among the concrete structures of the fusion reactor facility.
These impurities produced radioactive nuclides with gamma decay such as Co-60, Cs-134, Eu-152 and Eu-154. Especially Eu-152 and E-154 were the most significant radioactive nuclides produced from impurities of concrete because shutdown dose due to them is much higher than the other radioactive nuclides. The increased shutdown dose and induced activity of the concrete were quantified by the content of impurities.

P3 / 1134

P3.087 A quasi-3D thermal-hydraulic model for an HTS CroCo conductor

The HTS CrossConductor (HTS CroCo) was recently proposed by Karlsruhe Institute of Technology as novel concept for the winding pack of future fusion magnets.
The conductor concept is based on a cable-in-conduit configuration (CICC), in which 6 HTS CroCo macro-strands are twisted around a round copper core, jacketed in a stainless steel conduit and cooled by forced-flow supercritical helium at 4.5 K. Each HTS macro-strand is composed of twisted and stacked REBCO tapes surrounded by a copper sheath. Similar conductor configurations are cur-

rently under development and testing at other research laboratories.
Due to the geometry and to the thermo-physical properties of the materials involved, i.e. very low thermal conductivity of the HTS tapes with respect to the surrounding copper, non-negligible temperature gradients are expected within the superconducting region during relevant transients, e.g., quench propagation. Therefore, while a mono-dimensional treatment of the coolant flow and the jacket along the CICC could still be acceptable, there are good principle reasons to doubt about the applicability of well-established 1D thermal-hydraulic models, used so far in LTS conductor (and coil) analyses, to the HTS macro-strands.
Based on the time scales of the phenomenon under investigation, two different approaches are proposed here in order to capture the transverse temperature gradients within the HTS region of each macro-strand. For slow transients, e.g., the plasma burn, the HTS and copper can be separately approximated by a 1D model along the macro-strand direction, thermally coupled through a 1D radial model of the HTS. For faster transients, e.g., quench propagation, a 2D axisymmetric model along each macro-strand, with a thermal coupling between neighboring macro-strands, is proposed. After suitable verification, the reliability of the two developed quasi-3D models (1D and 2D along each macro-strand) is benchmarked against a detailed full 3D model.

P3 / 1048

P3.001 Quasi-optical polarizer system for ECH/ECCD experiments in the QUEST spherical tokamak

A transmission line has been newly developed in QUEST spherical tokamak (ST) for the highly efficient electron cyclotron heating and current drive (ECHCD) experiments with a 28 GHz gyrotron. Waves in plasmas can be described in terms of two eigenmodes, so-called extra-ordinary/ordinary modes, coming from anisotropic property of the electron motion on the magnetic-field direction. The modes have different polarization states, so they can be excited through the incident polarization control. The ECHCD effect depends on the incident mode or the polarization state of the propagating beam. All elliptical polarization states can be controlled in combination with two corrugated directions of the plates with respect to incident planes of the waves. The two corrugated groove depths are one-quarter and one-eighth wavelength, respectively. The two corrugated-plates were designed, and then were fabricated with careful attention to reduce Ohmic losses by means of high-precision milling. Arcing events were frequently detected at the polarizer section when the high-power millimeter-wave were transmitted. A new quasi-optical (QO) concept for the polarizer system is proposed to avoid the arcing here. In the QO polarizers, a coupling phase-inversion mirror was designed with the based on the Kirchhoff integral. The new QO polarizer has three components (two corrugated-plates and one coupling phase-inversion mirror). To check the polarizer performance, a low-power test system on heterodyne detection has been prepared. The amplitude ratio and the phase difference between horizontal- and vertical-fields of intermediate frequency at the heterodyne detection were measured with a network analyzer, and then were used to evaluate the polarization states. The evaluated polarization states are comparing with those calculated by the numerical code. The QO system with the two corrugated-plates and one coupling phase-inversion mirror will be introduced, and the evaluated and calculated polarization states will be discussed.

P3 / 1056

P3.009 Fast scenario design for EAST alternative magnetic diverted discharge

Advanced magnetic divertor configuration is one of the attractive methods to spread the heat fluxes over divertor targets in tokamak because of enhanced scrape-off layer transport and an increased plasma wetted area on divertor target. Exact snowflake (SF) for EAST is only possible at very low

plasma current due to poloidal coil system limitation. EAST can be operated in quite flexible plasma shapes like double null (DN) or signal null (SN) divertor configurations, but the external poloidal coils location was not optimized for the the snowflake. However, an alternative magnetic diverted configuration [1], characterized by two first-order X points where one is located in the primary separatrix and the other is outside the vacuum vessel, can be optimized to satisfy EAST constraints, like maximum current on poloidal field (PF) coils. In order to build the alternative magnetic diverted discharge in EAST, a fast scenario design tool based on EFIT & F2EQ code has been developed. A series of coil currents can be calculated and optimized based on a given alternative magnetic diverted configuration and desired pulse length. These coil currents will be offered to plasma control system (PCS) as feedforward coil currents. Then desired alternative magnetic configuration will be achieved at the scenario with specified time slice with the help of the designed feedforward and the feedback coil currents by RZIP or ISOFLUX algorithm control. This scenario design tool can provide a fast way to operate a new configuration discharge without using time-consuming TSC code.
[1] G. Calabrò, B.J. Xiao, S.L. Chen, et al., Nuclear Fusion, 55 (2015) 083005.

P3 / 1061

P3.014 Pre-qualification of the ELM resilient long pulse WEST ICRF launchers in the TITAN testbed

Before their installation and commissioning in the tokamak, WEST ICRF launchers undergo two categories of pre-qualifications tests. These tests aim at accelerating the commissioning of the launchers in the tokamak.
The first category of tests is milliwatt-range radio-frequency (RF) experiments which allow checking the launchers coupling capabilities, impedance matching and their load-resilience. See detailed in accompanying paper (W. Helou, et al., this conference).
The second category of tests is performed in the TITAN vacuum chamber and is detailed in this article. In particular, the adopted procedures for the latter tests are described. The TITAN vacuum vessel is designed to welcome WEST ICRF launchers, but also to provide ITER relevant vacuum condition, with dimensions compatible with 1/4th of the ITER antenna. This equipment is composed by a tank, pumping system, baking system.
When installed in the vacuum chamber, the launcher undergoes at first thermal cycling experiments. These experiments aim at testing the vacuum tightness of the full assembled launcher. In particular the tightness of all water cooling circuits is checked. This step is mandatory prior the installation in the machine, in order to avoid any leak during experiments.
Once the vacuum tightness validated, the launcher undergoes high RF voltage experiments. In these experiments the RF voltages and currents at the launcher's front-face are raised to the values corresponding to the plasma operation (27 kV and 915 A). Hence, these high RF voltage experiments allow validating the voltage standoff of the launchers. A dedicated process is used to increase step by step the voltage and duration of RF pulses. A description of this process is included. At the same time, the security systems, and in particular, the various arc detection systems are commissioned. All these procedures used by staged approach, will be discussed within the ITER ICRH team for their own antenna.

P3 / 1066

P3.019 42GHz ECRH system on Aditya Upgrade

The Aditya tokamak has been upgraded to Aditya-U by changing it's vacuum vessel from rectangular to circular to accommodate the diverter coil for shaped plasma. The tokamak has been commissioned and now operating with routine plasma experiments.
The 42GHz ECRH (Electron Cyclotron Resonance Heating) system has been integrated with the

tokamak. The system is capable to deliver 500kW power for 500ms duration. The microwave source Gyrotron is installed in SST-1 tokamak hall and ~75 meter long corrugated waveguide based transmission line is laid to launch the power in Aditya-U. The ECRH system with modified layout has been integrated successfully; the burn pattern close to tokamak port ensures good Gaussian beam (mode purity > 95%) at the exit of line.
Earlier the Aditya Tokamak used to operate at lower magnetic field (Max. 1.0Tesla) and the ECRH assisted low-loop voltage plasma start-up experiments were carried out at second harmonic at variable magnetic field (0.75T to 0.85T). The normal loop voltage for Aditya during start-up is around 22V, while the successful plasma start-up assisted with ECRH has been achieved at as low loop voltage as 6V.
Now the upgraded Aditya is operated at higher magnetic field up to 1.5T. So the ECRH experiments will be carried out at fundamental harmonic. This will reduce the requirement of EC-power as the power absorption is better at fundamental harmonic. The EC power in ordinary mode will be launched from low field side and EC-assisted low-loop voltage plasma start-up and off-axis plasma heating experiments will be carried out at fundamental harmonic (magnetic field : 1.3T to 1.5T).
The paper discusses about ECRH system on Aditya-U. The detailed paper will include the latest results of ECRH assisted breakdown and Heating on Aditya-U. The second harmonic EC-assisted low loop voltage plasma start-up in Aditya will also be presented.

P3 / 1072

P3.025 The heating systems capability of DTT

The purpose of the italian Divertor Tokamak Test (DTT, 6 T, 5.5 MA, R0 = 2.08 m, a = 0.65 m for ~ 100 s) is to study power exhaust and divertor load in an integrated plasma scenario. To accomplish this mission DTT will be equipped with 45 MW of additional heating power to fulfill a PSEP/R ≥ 15 MW/m studing alternative divertor configurations in view of ITER operations and DEMO design. The heating systems considered are NNBI (Negative Neutral Beam Injector), ECH (Electron Cyclotron Heating) and ICH (Ion Cyclotron Heating). Power repartition is based on the physical and functional requirements resulting from the review of the initial design [1]. The ECH is based on the use of 170GHz/1MW gyrotron, exploiting the development done for ITER and the experience of W7-X EC system. To fulfill major requirements assigned to EC waves 20-30 MW at plasma are foreseen. The NNBI system will be based on two beamlines providing deuterium negative ions (D-) at 300 keV and an injected power of 5-8 MW each. To increase efficiency and controllability, a modular approach will be adopted for the ion source and accelerator.
The main task of the ICH system is the central deposition in different heating schemes (mode conversion, minority heating and the novel three ions scheme). The frequency range is 60-90 MHz with a coupled power of 5–10 MW depending on the adopted antenna.
A high reliability in delivering power to plasma will be obtained considering consolidated and available technologies together with a design architecture capable to reduce the fault risks. An overview of the three systems will be given, with a particular attention to the design integration in the DTT plant and to synergetic aspects between them.

[1] F. Crisanti et al. FED, Vol 122, Nov 2017, 288-298

P3 / 1099

P3.052 UWAVS first mirror after long plasma cleaning: surface properties and material re-deposition issues

One of the important aspects of the plasma cleaning of the front-end mirrors (FM) in ITER UWAVS diagnostics is to understand surface roughness after multiple cleaning runs and to minimize possible contamination due to unwanted sputtering of the mirror surface and neighboring walls. The capac-

itive coupled RF 30-60 MHz is a candidate for the UWAVS FM cleaning. It generates ion fluxes of tens of electron-volts sputtering contaminants as well as construction materials. In the present report, we discuss materials analysis with electron diffraction spectroscopy and x-ray photo-electron spectroscopy for Al, Mo and W samples after cleaning with 30-60 MHz discharges in He for various locations inside the vacuum compartment. Typical ion energies were 50-70 eV at the RF electrode and 20-30 eV at the grounded wall. Mirror reflectivity was measured after removing W, Al, Al2O3 contaminants in 60-hour exposure and showed satisfactory results. Materials and their oxides sputtered from the FM were deposited at distances 2-4 cm from the electrode with rates 0.1-0.05 nm/hr. At the Second Mirror, the deposition rates were estimated 0.01 nm/hr. or lower. The experiments showed advantages of Mo a construction material due to lower sputtering. Dielectric material traces were found due to electrical damage of cable feedthrough at higher powers. Measures to prevent the sputtering of dielectrics are discussed. The Mo-mirror was etched at 0.5-1 nm/s rate. That did not change the mirror reflectivity after 100 hours of exposure, but the overall effect on the reflectivity considering full service time is yet to be investigated.

P3 / 1102

P3.055 LIBS system on linear plasma device PSI-2 for in situ diagnostics of plasma-facing materials

One of the keen interests of plasma-facing material diagnostics is the in situ measurement of fuel content on the material surface and monitoring the outgassing process in the early stage after plasma exposure. The W samples are exposed to D plasma (with a typical fluence of 5×1025 D/m2) on the linear device PSI-2. Laser induced breakdown spectroscopy (LIBS) is employed as the detecting technique thanks to its ability of in situ measurement [1]. In our previous work [2], the measurement starts at 40 minutes after plasma exposure due to the optical adjustment required prior to each measurement. In the current work, effort has been made to reduce the delay before the measurement.

The current LIBS system is based on an existing INNOLAS Q-switched Nd:YAG laser (450mJ in 6ns, 532nm, 10Hz) and has been established on the plasma exposure chamber on PSI-2. The measurement starts at 0.1 − 1 second, which is a significant improvement to the previous measurement. This laser can also be used for postmortem analysis when the optical path is adjusted to the exchange chamber.

The LIBS results are also compared to TDS and NRA measurements. The TDS and NRA were performed 2-3 weeks after the sample was exposed in PSI-2. The TDS result showed that the remaining D on W surface was in the range of 5×1016 D/cm2. The NRA result showed that most of the D was detected within the depth of 1μm. The postmortem LIBS signal was 1000 counts (Dα line) per pulse in a crater of Ø1 mm×25nm. This gives an early estimation of 1014 to 1015 D/LIBS measurement. Improvement of the LIBS detection sensitivity is undergoing and further calibration using the lab sphere will be used as a reference.

P3 / 1125

P3.078 Strain distribution in the Nb3Sn Rectangular Wind & React Conductor of the European DEMO project determined by inductive measurements

Cable-in-Conduit Conductors (CICCs) are complex systems whose behaviour is not directly predictable studying their single components, and the explanation of their observed properties is not straightforward. The knowledge of the strain (Eps_th) distribution of Nb3Sn filaments in a CICC cross-section is a key parameter in understanding the performance and its evolution when the cable

undergoes electromagnetic cyclic loading. This work focuses on the Wind and React (WR1) rectangular CICC sample designed by ENEA for the European DEMO project, that has exhibited a current sharing temperature (Tcs) above 6.9 K at 13 T and 81.7 kA, during its qualification tests in the EDIPO facility at the Swiss Plasma Centre. Compared to the critical current tests on the constituting wires, the observed CICC performance corresponded to an effective strain value of -0.55%. This work aims to interpret this result on the basis of the analysis of the strain distribution inside the cable itself. Susceptibility versus temperature, Chi(T), of WR1 constituting free standing strands has been measured before and after etching of the stabilizing copper, and it has been compared with the susceptibility measured on the cable during the test campaign in EDIPO. A well-known and already assessed method has been used, starting from susceptibility measurements, to compute the Tc distribution in the cable cross-section and, subsequently, to infer from that the thermal strain distribution. The obtained results give an indication that the Eps_th distribution in WR1 cable is narrow, compared to the strain distribution calculated with the same method for most ITER cables.

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P3.804 Steady-state transverse heat transfer in a single channel CICC

Current models used for thermal–hydraulic analyses of forced-flow superconducting cables, used in the fusion technology, such as, e.g. Cable-in-Conduit Conductors (CICCs), are typically 1-D. They require reliable predictive expressions for the transverse mass-, momentum- and energy transfer processes between different cable components, in order to reliably simulate the behavior of any superconducting magnet design either in normal operating conditions or during a quench evolution. Only few heat transfer correlations for flow in a CICC bundle have been proposed in the literature, but none of them is widely accepted for predictive purposes. As a result, in thermal-hydraulic studies of conductors designed for the EU-DEMO coils standard heat transfer correlations for flows in smooth tubes are used, which are undoubtedly over-conservative in this case. Systematic measurements of heat transfer coefficients in a CICC bundle should be performed to provide an experimental database for further attempts to develop a predictive correlation. In 2017 we performed first preliminary measurements of the steady-state heat transfer coefficient between a jacket wall and fluid flowing (i) in a smooth tube and (ii) in a CICC bundle of the reference sample (JT60SA TF conductor). Based on these preliminary results, recently we improved the experimental configuration, and we performed further systematic measurements of the heat transfer coefficient at various values of the water inlet temperature and mass flow rate, to obtain the results in a wide range of Pr and Re numbers. We also made some attempts to improve the heat transfer model used in the experimental data analysis. These new results are discussed in the present paper.

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P3.094 Validation of the updated DTT TF coil conceptual design

The Divertor Tokamak Test (DTT) facility is a satellite experiment in the same research and development framework of the European DEMOnstrating fusion power reactor. It shall evaluate different divertor solutions for power and particles exhaust, and shall investigate the plasma-material interaction scaled to long pulse operation. It is part of the European Fusion Roadmap and shall be completed in Italy. ENEA is designing the complete magnet system and is performing the great part of the mechanical finite element analyses. The design of the machine has been reiterated from what initially presented in 2015, as the major plasma radius has been reduced from 2.15m to 2.08m. The analyses hereby reported are thus aimed to support the validation of the updated design by focusing on the response of the Toroidal Field (TF) coil system. Mechanical stress resulting from the magnets' cool-

down and from electromagnetic loads due to different operative conditions have been computed. The results are a combination of bi- and tri-dimensional analyses. The first make use of the generalised plane strain formulation to model in full detail the equatorial section of the TF inner leg. The latter include the complete TF coil system and implement homogenised material properties for the Winding Pack (WP). Particular attention has been put in correctly modelling the structural contacts and the wedging effect of the TF coil. This report thoroughly describes the developed approach. The presented results show that stress concentration values are below allowable limits. Hence, this work validates the latest design update of the DTT TF coil system.

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P3.095 Mechanical Analysis of the DTT central solenoida

The "Divertor Tokamak Test" facility, DTT, is a project for an experimental tokamak reactor developed in Italy, in the framework of the European Fusion Roadmap.
In this design phase of the machine it is necessary to ensure the structural integrity of the superconducting magnets.
This work focuses on the analysis of the stresses that are generated in the central solenoid of the tokamak, the CS system, a coil composed of 6 stacked and independently energized superconducting modules, wound from Nb_3Sn Cable In Conduit Conductors, working in a pulsed regime.
The peak of magnetic field reached on the coil is about 13.5 T and the whole structure must be cooled down to 4.5 K, operating temperature in which the cables work in the superconducting phase.
The electromagnetic loads generate a field of Lorentz forces on the magnet with predominantly centrifuge action, leading the coils to open and counteracting the contraction effect due to the cooling of the magnet.
It is very important to evaluate the stress behavior in the jackets of the superconducting cables, which perform the main structural support action and perform the containment of the refrigerant helium circulating in the cables, and the insulation of the coil.
Therefore, an FEM model has been defined to carry out a numerical simulation. The model includes full detail of the axisymmetric section of the coil, including the superconducting material, the 316-LN steel of the jacket and the filler and insulating resin with their specific properties.
The model also considers interactions and exchanges of forces between the 6 CS modules, thanks to the implementation of the contact elements.

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P3.106 Ultrasonic analysis of tungsten monoblock divertor mock-ups after high heat flux test

In a fusion reactor, heat exhaust is one of the most challenging engineering issues, due to the high heat flux (HHF) expected on the divertor targets. The tungsten (W) monoblock design represents one of the most suitable technological solution for plasma facing components, since it has already met the ITER requirements. However, further research is required to investigate improved solutions to withstand DEMO specific loading. Among the crucial aspects, embrittlement due to neutron irradiation and component life-time estimation play a central role. For this reason, alternative W-monoblock divertor concepts have been developed and tested under HHF cycled loading, in order to assess their performance and expected lifetime under DEMO relevant heat loads and cooling conditions. To support such campaign, post-test non-destructive analyzes, such as ultrasonic test (UT), are required to quantify the damage experienced by the components. In fact, UT allows detecting defects and their location inside the material with good accuracy since it relies on the propagation of mechanical waves which is very sensitive to discontinuities.
In this contribution, a comparison between post-HHF-test UT measures of different W-monoblock

concepts is presented. The baseline reference of water-cooled divertor targets (ITER-like) is compared with two other concepts, in which the interlayer between monoblocks and tube has been engineered to increase the performance of the component (Thermal break interlayer, Thin graded interlayer). For each concept, a respective mock-up is analyzed and the ultrasonic measures are reported at the depth of interest. Scans are shown to display the reflected signal amplitude and time of flight at defined depths, therefore giving insight to the shape and entity of the defect, when present. The results will be reported with particular attention to highlight the differences both in the defect morphology and in their location, in particular for defects in critical regions of the component.

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P3.110 Temperature dependence of hydrogen co-deposition with metals

Hydrogen co-deposition with sputtered particles is one of the main channels of hydrogen isotope accumulation in today's tokamaks. According to experiments in tokamaks and in labaratory conditions hydrogen concentration in co-deposits show that can be very high (up to tens of atomic percents) for various materials at low deposition temperature even in the case of low hydrogen solubility. This is explained by presence of a high concentration of defects in the deposited films, and therefore, temperature dependences of hydrogen accumulation in co-deposits are very different for various materials due to difference in energy characteristics of defects in materials.
In the present work, we propose a model of H co-deposition with metals and an analytical formula for H/Me ratio in the deposited film. The model predictions were compared, firstly, with experimental results on W-D co-deposition [1], where W-D co-deposited layers were produced by means of DC magnetron sputtering of W target in Ar/D working gas mixture with the substrate temperatures T varying from RT to 800 K with the step of 30 K. The deposited layers were analyzed by means of in-vacuo thermal desorption spectroscopy (TDS) with no contact of the sample with air. The experimental dependencies of D/W vs T had several steps of D content decrease, and the proposed model described the experimental features well in all the temperature range used.
Similar experiments were repeated for two other materials: molybdenum and aluminum. The experimental results were also well described by the model.

[1] S.A. Krat et al, Vacuum, 149 (2018) 23-28

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P3.112 High heat-flux response of high-conductivity graphitic foam monoblocks

Plasma-facing components based on so-called monoblocks are planned for use in the divertor region of long-pulse plasma devices such as ITER and JT-60SA due to their capacity to handle high heat fluxes with active water cooling. The plasma-facing materials that are preferred for these monoblocks are tungsten for ITER or carbon-carbon fiber composite (CFC) for JT-60SA. The requirements for the plasma-facing components include the ability to handle high plasma fluxes resulting in high temperatures. Therefore, reasonably high thermal conductivity is required. In this study, graphite foam is explored as a monoblock material. Graphite foam is commercially available, is virtually isotropic (as opposed to the orthotropic CFC), and has a room temperature thermal conductivity as high as 285 W/m-K. Plasma compatibility has previously been demonstrated with small samples in the PSI-2 linear plasma device in Juelich, Germany, and the Wendelstein 7-X (W7-X) stellarator in Greifswald, Germany. The graphite foam monoblocks used for this study are manufactured and joined to CuCrZr tubes, both by brazing and by a simple press fit. The test articles

consist of nine graphitic foam monoblocks 28mm x 28mm x 28mm joined either to a single 12-mm-OD CuCrZr tube or to two parallel 10-mm-OD CuCrZr tubes. These monoblocks are then tested in the GLADIS high heat-flux facility in Garching, Germany, with heat fluxes up to 10-15 MW/m2 and thermally cycled in order to examine robustness for long-pulse divertor applications. Experimental results are compared with computational fluid dynamics simulation. This new technology is under consideration for the proposed actively cooled W7-X divertor scraper element.

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P3.114 Instrument for hydrogen isotopes irradiation and resident assessment

Tritium takes the most cost of fusion project when it has been regularly operated.In order to give a proper fuel combustion rate and recycling efficiency,it is necessary to assess the amount of hydrogen isotopes(accompanied with helium)retent in the plasma facing materials(PFM).
A comprehensive ECR plasma system for tritium (named CEPT)is designed and built for the assessment of tritium penetrating and resident in PFM(e.g. tungsten).ECR(electron cyclotron resonance)plasma can provide low pressure,high density,long-term stable irradiation of hydrogen isotopes.The typical irradiation parameter of hydrogen isotopes or helium plasma produced by CEPT is:3eV-7eV,1020D/(m2s)-1021D/(m2s),10-3Pa-1Pa, 0-100sccm.A farther 600V bias electric field can effectively accelerate the ions.Contrasting to the unstable arc discharging(which can provide higher parameters of ion flux but short-term stable working due to electrode material loss, higher fuel consumption) ,CEPT can meticulously produce hydrogen isotopes and helium flux on the PFM more than 12h continuously. CEPT mainly has a glow vacuum chamber, a plasma irradiation vacuum chamber, and a TDS (Thermal Desorption Spectroscopy)-LIBS (Laser Induced Breaking-down Spectroscopy) vacuum chamber. All of them can obtain 5.5×10-5Pa ultimate vacuum. The material samples can be water-cooled when facing the plasma irradiation.And the hydrogen isotopes resent in the sample can be analyzed online. The magnet field of the confined plasma is adjustable up to 2500 Gauss (water cooled coils).A Langmuir probe is used to diagnose the plasma density and energy.The radiation protection system was designed to ensure the safety of tritium operation.

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P3.125 Life time estimation of ITER first wall mock ups by FEM simulations

The ITER first wall (FW) panels consist of plasma facing Be tiles, the CuCrZr alloy as heat sink material, and the stainless steel as structural material. A copper layer of 1~2 mm is used between the Be tile and the CuCrZr for stress compensation. Cyclic high heat flux tests employing the electron beam facility results indicate that the failure/weak spot usually occurs at the joint corners between the Be tile and the copper layer. 3-D FEM thermo-mechanical analyses utilizing ANSYS software have been conducted to detect the failure mechanism. Within the ANSYS strain-life fatigue module, based on strain-life relation of copper from ITER materials handbook, the fatigue life prediction of 3 ITER FW mock up (MU) types was performed. The 'signed' equivalent stress, which reflects the real cyclic load spectrum of the material, was chosen for the stress component for the fatigue life prediction. Considering the singularity issue, the location of 50 micrometer apart from the free surface was selected as the investigation spot. The results show that, when the MU is loaded with 2 MW/m^2, the semi-prototype (Be tile size of 46*36*10mm3), the small size MU (Be tile size of 36*36*10mm3), and the small size MU with cassette design have the fatigue lives of 10834 cycles, 9773 cycles, and 12000 cycles, respectively. It can be seen that the size of the tile as well as the design of the MU affect the lifetime of the MU. It can also be seen that, all our predicted life times are below the design criterion for the ITER FW MUs (15000 heat load cycles). Discussions on the tile size effect, stress

component selection, as well as the singularity affection on the fatigue life prediction are presented in this paper.

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P3.139 Real-time volumetric rendering of radiation fields using 3D textures

The operation of nuclear fusion facilities must be carefully planned and monitored due to the potential damage to equipment or personnel caused by radiation fields. A method for visualising such three-dimensional (3D) radiation fields in real-time is presented. An interactive volumetric representation is achieved using view-dependent ray casting of a scalar field in three dimensions.

Real-time performance is achieved by exploiting parallel processing on the graphics processing unit, with scalar field data uploaded as 3D textures. Lack of consumer hardware support for true 3D texturing is mitigated with software indexing of two-dimensional textures containing multiple XY planes through the volume. Integration with a commercial-off-the-shelf software framework, Unreal Engine 4, allows fusing with context geometry derived from computer-aided design engineering models and viewing with virtual reality headsets.

We present a novel technique for visualisation of 3D field data in real time. While targeted at rendering radiation fields for remote maintenance planning in nuclear facilities, the method inherently generalises to the visualisation of any scalar quantity that can be sampled in a three-dimensional field, such as neutron dose, plasma density, temperature or pressure. Qualitative results are presented and future extensions discussed.

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P3.141 F4E Load Transfer Procedure among Finite Element Models different in Topology or in Discretization

In magnetic confinement fusion ITER represents the most challenging projects conceived ever. The assessment of ITER structural behavior is not trivial since it requires the application of loads coming from different types of analysis (electromagnetic (EMAG), thermal, dynamic, etc.), which are usually run using different software and Finite Element (FE) models, onto mechanical (MECH) models representing the system itself. The load interpolation between meshes dissimilar in discretization or in topology needs to be carefully performed in order to avoid that such an incompatibility could affect the global or local mechanical equilibrium of the system, not correctly preserving the spatial distribution of forces and related moments.

In this paper, a methodology developed in Fusion for Energy (F4E) for interpolating mechanical loads both between compatible (i.e. from solid to solid models different in discretization) and incompatible (e.g. from solid models to shell/beam models) FE models will be described in detail. This novel procedure is able of transferring a force vector field (i.e. Lorentz forces) from a three-dimensional solid mesh (i.e. the EMAG model) onto a target mesh (i.e. the MECH FE model), being it either three-dimensional solid or simplified beam/shell model.
This interpolating procedure is developed with the aim of preserving both the global and the local mechanical equilibrium of the system in terms of resultant of forces and overturning moments.
The quality assessment of this procedure is based on the comparison between the global force and moment resultants of the source and the transferred load field. A few examples and their related results will be discussed in support of this methodology. Furthermore, an exhaustive application of this procedure can be found in the paper [1].

References:

[1] Edoardo Pompa, Gabriele D'Amico, Pietro Testoni, Didier Combescure, First Wall #6 mechanical assessment using the F4E Load Transfer Procedure for electromagnetic loads distribution.

P3 / 1194

P3.147 Thermo structural analysis of the in-vessel components of the ITER plasma position reflectometry system on the high field side

The Plasma Position Reflectometry (PPR) diagnostic systems, to be installed in the International Thermonuclear Experimental Reactor (ITER), will measure the edge electron density profile of the plasma, providing real-time supplementary contribution to the magnetic measurements of the plasma-wall distance. Some of the diagnostic components will be placed inside the vacuum vessel (VV) and therefore directly exposed to the plasma, subjected to high radiation doses which will contribute to increase the thermal loads in the PPR system and may cause irradiation induced changes in the material properties. The thermal loads may cause excessive temperatures on some of the PPR in-vessel components compromising their structural integrity. Therefore, the main objective of this work was to estimate the temperature distribution on the PPR in-vessel components and the associated thermal stresses in order to assess their structural integrity. All PPR in-vessel components are made of austenitic stainless steel 316L(N)-IG.
For this purpose, finite element models of the PPR in-vessel components of gap 6 were developed, using ANSYS V18.2, based on the most up-to-date CAD models of the PPR in-vessel components available in ENOVIA. These CAD models were then analysed, using CATIA R23 and ANSYS SpaceClaim V18.2, to mitigate inaccuracies and clashes. Afterwards, the thermal loads were applied and steady-state and transient finite element analyses conducted. The thermal stresses were then determined by carrying out structural analyses. The results of the analyses are presented and discussed from the point of view of the structural integrity of the components.

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P3.149 Design and manufacturing of the wendelstein 7-X cryo-vacuum pump

The cryo-vacuum pump (CVP) system, consisting of 10 units distributed symmetrically inside the Wendelstein 7-X plasma vessel, will be installed together with the 10 units of the actively cooled high heat flux divertor. One pump each is located below the corresponding divertor, and positioned as close as possible to the flux line strike points in order to allow efficient control of plasma density and for screening impurities. Each CVP is divided into two parts, interconnected by a transfer line, to ensure access for divertor diagnostic integration. The CVP panels are operated with supercritical helium at 3.3-3.8K to pump discharge gases such as He, H2 and D2. They are protected against thermal radiation by black-oxide finished stainless steel chevrons operated with liquid nitrogen at 77K. In front of this LN2-shield is a water-baffle which protects the CVP against plasma and ECRH stray radiation. It consists of copper chevrons with zero overlap, mounted on a water-cooled steel pipe. These chevrons are coated with a Al2O3-TiO2 layer. In addition, an uncooled copper shield covers the gap between water-baffle and LN2 chevrons in order to prevent stray radiation to take this path to the He-cooled panel. The cryogenic fluids are supplied via a dedicated port plug-in which is thermally insulated by a LN2-cooled cryo shield and superinsulation. All 10 CVPs have already been manufactured and successfully leak tested under hot and cold conditions in the workshops of IPP Garching.
This paper presents the design and the manufacturing technology of the CVPs and the adjacent periphery. Results of the quality assessment such as integral He leak testing at 160℃ and cryogenic temperatures (-196℃) are also discussed.

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P3.152 WEST regular in-vessel inspections with the Articulated Inspection Arm robot

Currently on fusion devices, diagnostics are mainly aiming at plasma analysis and control. However, operational and programmatic needs have appeared for regular in-vessel components monitoring during plasma campaign. Light robotics systems could meet this requirement and may be a way as well to replace human interventions to fix damaged in vessel components. To minimize the impact on machine operation, the robotic system has to be mini-invasive and compatible with operating conditions (vacuum, temperature, radiation…).

To fulfill this goal, CEA has developed a multipurpose carrier able to be operated inside WEST vessel between plasma pulses. A prototype of this robot, called Articulated Inspection Arm (AIA), was tested in 2008 in Tore Supra vacuum vessel. A major upgrade was performed in 2014-2015 with the aim of converting this prototype into a reliable tool in support to WEST operation. During the WEST components manufacturing and installation (2014-2016), the robot was integrated and tested in the EAST Tokamak.

Since 2017, the AIA has been regularly used during the WEST plasma campaigns. Movies provided by the embedded camera allow to assess the evolution of Plasma Facing Components surface state and the effects of plasma loads, runaways and disruptions.
The robot operation was also very helpful to assess the needs for maintenance, to assist mechanical assembly without man entry and to perform diagnostics calibration under relevant conditions.

The paper will detail lessons learned from the robot integration and use on the WEST Tokamak. Future developments for innovative embedded diagnostics will also be presented.

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P3.155 Assembly methodology and results for WEST platform PFCs positioning

Since 2013, CEA has carried out an in-depth modification of the Tore Supra tokamak to build the WEST platform, targeted at supporting the ITER tungsten divertor detailed design, manufacturing and operation. The changes included the modification of the magnetic configuration with new in-vessel coils, the replacement of all carbon Plasma Facing Components (PFCs) by new tungsten elements and the upgrade of the radio frequency heating systems as well as plasma diagnostics.

One of the technical challenge of this tokamak transformation consisted in assembling and accurately positioning interlinked new WEST components (supports, coils, PFCs, thermal shield panels, embedded diagnostics…) in an existing machine with numerous vessel geometric imperfections and asymmetries. All the PFCs structures were located in the magnetic referential system with a sub-millimeter accuracy and the misalignments between neighboring divertor sectors were kept under 0.3 mm, i.e. in the relevant tolerance requirements presently considered for the ITER tungsten divertor. At the same time, the volume allocated for the plasma was enlarged thanks to a minimization of the distance between coil structures and the vacuum vessel.

The paper will summarize the work performed during WEST assembly phases and will present the key technologies and methods that were used to reach this level of precision in the components integration. It will describe both the design principle applied for tailoring interfaces between components and the advanced metrology technics used. The specific handling tools developed to accurately position large frames (up to 10 tons for both upper and lower divertor structures) and the kinematic modelling performed to prepare and check assembly trajectories will also be presented. Finally, metrological results will be discussed with a focus on component alignment achievements with regards to ITER PFCs assembly needs.

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P3.169 Provisions for a cryogenic distillation process linked to a CECE process at ICSI Rm. Valcea

Nuclear reactors whether they are based on fusion (JET, ITER, DEMO), fission (e.g. CANDU type), or cooled using molted salts (MSR's) generate significant amounts of wastes in the form of low level tritiated light water or heavy water, for which there is an increasing interest to process and recover tritium (in gas form) and deuterium (as heavy water). Current water treatment systems allow the recovery of heavy water above 1% D2O which leads to a considerable loss of heavy water from a CANDU type reactor.
Provisions are being made in ICSI to design and build an experimental facility based on CECE process coupled with a CD system to process low level tritiated heavy/light water for reactors that generate these wastes in view of increasing the recovery factor of tritium, material which is of great interest for the commissioning and start-up of fusion nuclear reactors.
This paper aims to make some provisions as to design and mathematically simulate the CD process which is to be connected to a CECE process and provide with some results based on the design and simulation model in support for the development of such detritiation technology.
Based on the input data (e.g. the flow rates, the composition of the gas supplied into the cryogenic distillation cascade, temperature and pressure drop along the column, liquid inventory) the design and simulation model will provide the distribution of all the molecular species involved along each cryogenic distillation column, the temperature, pressure, vapor and liquid enthalpies profiles along the columns and products of each column.

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P3.171 Addressing the feasibility of inboard direct-line injection of high-speed pellets for core fuelling of DEMO

Core fuelling of the EU-DEMO tokamak is under investigation within the EUROfusion Work Package "Tritium, Fuelling and Vacuum". Pellet injection still represents the most promising option. Modelling of pellet penetration and fuel deposition profiles for different injection locations, assuming specific DEMO plasma scenarios and the ITER reference pellet mass (6×1021 atoms), indicates that effective fuelling can be achieved launching pellets from the High Field Side (HFS) at velocities ⊠⊠1 km/s. To implement suitable inboard injection schemes for DEMO, two complementary approaches are being considered: one uses guide tubes with curvature radii ⊠ 6 m, to preserve pellet integrity at ⊠ 1 km/s; the other is investigating the feasibility of injecting high-speed (⊠ 3 km/s) pellets along "direct line of sight" (DLS) trajectories, from either the HFS or a vertical port. The identification and integration of suitable oblique HFS straight injection paths, not interfering with the central solenoid (CS), requires careful investigation. Options using the upper vertical port have been therefore explored first, as they represent the simplest approach; simulations using the HPI2 pellet ablation-deposition code indicate, however, that vertical injection may be effective only if pellets are injected from radial positions well inboard the port axis. High-speed injection through oblique inboard "DLS" paths, not interfering with the CS, are instead predicted to grant good fueling performance, provided that the trajectories intercept the separatrix at a distance from the equatorial mid-plane ⊠⊠ 2.5 m. The angular spread of high-speed free-flight pellets trajectories, and/or the suitability of straight guide tubes to reduce the corresponding open cross section on breeding blanket penetration, are being explored using an existing facility, in collaboration with Oak Ridge National Laboratory, with the initial goal of measuring the scatter cone of free-flight pellets. The neutron flux across DLS injection paths is also being investigated. Progress are reported.

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P3.173 V&V analyses of the GEANT4 Monte Carlo code with computational and experimental fusion neutronics benchmarks

The high-energy particle physics Monte Carlo code toolkit GEANT4 has been expanded for fusion energy-range neutron transport simulations based on evaluated nuclear cross-section libraries. Verification and Validation (V&V) analyses were conducted with nuclear data from the ENDF/B-VII.0 and the JEFF-3.1 library to show the suitability for fusion applications. Two computational benchmarks with a single fusion-relevant isotope at a time and one experimental benchmark were performed with GEANT4 in comparison with MCNP5.
The differential computational benchmark assumes a neutron beam with isolethargic energy distribution along the axis of a cylinder with length 2m and radius 1μm. After a single material interaction, the scattered or secondary neutrons passing through the cylinder side surface are recorded. The accumulative deviation between GEANT4 and MCNP5 throughout the energy spectrum is <1% for all isotopes and libraries.
The integral computational benchmark is a 30cm radius sphere with an isotropic 14.1MeV neutron source in the centre. The neutron flux averaged over the sphere volume is recorded. The total flux deviation between GEANT4 and MCNP5 is <1% for all isotopes and libraries with some larger deviations in individual energy groups for some isotopes.
The considered experimental benchmark, performed by the Institute of Physics and Power Engineering, Obninsk, is on the transmission of 14MeV neutrons through iron shells of different thicknesses (2.5 – 28cm). Geometry and source description were processed from the Shielding Integral Benchmark Archive and Database (SINBAD). In this experiment, neutron leakage spectra were measured at a distance of 679cm from the centre of the iron shells. The obtained Calculation/Experiment (C/E) ratios are acceptable for most of the energy spectra with a larger deviation in the range of 4 – 10.5MeV.
In summary, the results of the V&V analyses obtained in this work indicate the suitability of GEANT4 for the application to fusion neutronics problems.

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P3.178 Development of a SIMMER\RELAP5 coupling technique: a preliminary application to the LIFUS5/Mod3 test facility

The In-Box Loss Of Coolant (LOCA) postulated accident is considered as a major concern for the safety involving the development of EU-DEMO fusion reactor. Related to the renewed interest in the Water Cooled Lithium Lead (WCLL) blanket concept, a unique and innovative experimental campaign is under development at ENEA Brasimone research center aiming at investigating consequences of an In-Box LOCA event applied to the WCLL Breeding Blanket (BB). The LIFUS5/Mod3 experimental facility - and its related tests campaign -is aimed at investigating the PbLi/water chemical interaction after the occurrence of the postulated accident. Experimental data obtained from this LIFUS5/Mod3 experimental campaign are also of fundamental importance for the verification and validation of software codes applied to the deterministic safety analysis of the WCLL BB concept. In this frame, the University of Pisa developed a modified version of the SIMMER-III code implementing PbLi/water chemical interaction to account for the direct energy release in terms of temperature and pressure increases inside the BB.
In this paper, an in-house coupling tool between SIMMER and RELAP5 codes (modified versions implementing respectively PbLi/water chemical interaction and PbLi thermo-physical properties) is presented together with its preliminary application to the LIFUS5/Mod3 pre-test simulation. The coupling procedure can be classified as a “two way”, “non-overlapping”, “online” technique aiming at investigating multi-physics and multi-scales phenomena in support of the development of fusion technologies. The application of the 2D Thermal-hydraulic SIMMER code limited to the LIFUS5/Mod3 main vessel allows to simulate the chemical interaction between PbLi/water, while the water injection line - simulated by the RELAP5 code - supplies accurate boundary conditions to the SIMMER.

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P3.182 Effects of Pre-Dissociations of HTO and H2 on Tritium Recovery Rate of Purge Gas Flow in Tritium Breeding Blanket

Tritium recovery rate is one of most important parameters to design highly efficient fuel cycle in fusion reactors. To estimate the tritium recovery rate accurately, chemical reactions in the tritium recovery process must to be studied in detail. In solid type breeding blankets, tritium is expected to be released from the breeder pebbles in the form of HTO into purge gas surrounding the pebbles. The released tritium is then extracted mainly in the form of HT produced as a result of chemical reactions with H2 highly diluted in helium in the purge gas flow. In our previous study, we have studied the chemical kinetics of HTO+H2 -> H2O+HT reaction by the transition state theory. The results show that the activation energy of this reaction is relatively high, and the reaction rate is small for operational temperature ranges of Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM), adopted as a solid breeder blanket for the study, implying the necessity of further investigation of other reaction paths to produce HT efficiently.
In the present paper, the effects of pre-dissociations of H2 and HTO on the production of HT are studied by quantum chemistry. Firstly, the chemical kinetics of the dissociation of H2 and HTO is studied. Then, possible reaction pathways to produce HT are studied. Finally, the chemical reaction rate coefficients of the HT production reactions are presented.

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P3.184 Effects of Helium irradiation on hydrogen diffusion through erbium oxide coatings

To limit hydrogen leakages in a breeding blanket of fusion reactor, a hydrogen permeation barrier can be used. Erbium oxide was selected as a promising candidate with a low hydrogen diffusion. Thus, the purpose of this study is to understand the irradiation effect of helium ions, originating from fission of lithium exposed to fusion-induced neutrons in the blanket, on the hydrogen diffusivity of erbium oxide.
To meet this objective, an erbium oxide of approximately 600 nm thickness was deposited on the substrate with the same quality as blanket materials. After division into subsamples, halves were irradiated to 1–3 displacement per atom (dpa) with 0.4 MeV 4He+ from the 1MV Cockcroft-Walton accelerator. The irradiation conditions are chosen so as to meet mathematical models of lithium blanket, in which the stopping and ranges of ions in matters (SRIM) was used for the irradiation damage calculation. The samples were annealed at 773-973 K under hydrogen atmosphere (0.4 bar), as a safe substitute of hydrogen exposure. Through the nuclear reaction analysis of hydrogen, we will be able to obtain the depth profile of diffused hydrogen during the annealing period, from which information on the diffusivity for different samples is obtainable. Influence of irradiation on the hydrogen diffusion coefficient of erbium oxide could be quantified for non-irradiated and irradiated samples. Future experiments and analysis are now under way.
In conclusion, the rise of diffusivity due to the irradiation seems serious in blanket operation. In the next stage, the diffusion study will be required using hydrogen itself to assert the promising nature of erbium oxide, together with the structural analysis of irradiated erbium oxide.

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P3.186 Cryogenic distillation experimental stands for hydrogen isotopes separation

In cryogenic distillation columns complex phenomena appear, some of them are neededand othersmust be avoided, such the non-uniform cooling of the distillation column or the impossibility of transfer of the cooling power to the gases mixture with major changes in the separation dynamics. The loss of separation capacity or the inability to reach optimal operating parameters are caused by multiple factors starting with insufficient cryogenic power, heat losses from design/construction or from improper design/installation of the isolation systems, condenser with small heat transfer surface or cryogenic power, column misalignment and column internals not proper designed and installed, boiler affected by two-phase boiling and/or flow, feeding of the column with high flow and/or high temperature, improper position of the feeding and/or extraction points or improperpacking. The paper presents new solutions for the design of several components, parts of the new experimental cryogenic distillation stands developed in the Cryogenic Laboratory in order to expand the experimental base for ICSI Rm.Valcea "Experimental Pilot Plant for Tritium and Deuterium Separation". One stand will allow testing and characterization of new solutions for cryogenic distillation columns at laboratory scale and will be attached to hydrogen liquid liquefaction system while the second stand will allow testing of verified solutions at semi-industrial level and this will be attached to a Linde HRLS 11 refrigeration and liquefaction plant that has the capability to transfer in a closed loop helium gas at 16K-20K.

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P3.190 Lithium conducting ceramics for electrochemical sensors in molten metals

Lithium 6 is the isotope required to generate in-situ tritium in fusion reactors. Because of that, lithium monitoring in lithium-lead eutectic (Pb-15.7Li) is of great importance for the performance of the liquid blanket. Lithium measurements will be required in order to proof tritium self-sufficiency in liquid metal breeding systems. On-line lithium sensors must be designed and tested in order to accomplish these goals.
Solid state electrolytes have been successfully used for gas sensors in many applications. Sensors based on solid state electrolytes have several advantages: generally they are stable compounds which can withstand the harsh chemical environment of the melts, moreover its ionic conductivity increases with the temperature and the output signal (cell potential) is easy to measure.
Lithium conducting electrolytes for molten metals are under development at the Electrochemical Methods Laboratory at Institut Quimic de Sarria (IQS) at Barcelona. Its qualification and performance are being tested. Li-probes for molten metals will be based on the use of ceramic type solid state electrolytes.
In the present work, Li6BaLa2Ta2O12 and Li6.55La3Zr2Ga0.15O12 were synthesized and characterized. The solid state electrolyte was properly shaped in order to be used as a lithium probe. The sensor response was measured electrochemically in different molten lithium alloys.

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P3.191 Method of determination and optimisation of the control parameters for an LPCE process

The hydrogen isotopes separation plants have special requests related to safety operation and avoidance of radiological fluid leakage and explosion conditions. For the LPCE, part of the ICSI Rm.Valcea "Experimental Pilot Plant for Tritium and Deuterium Separation", the process transformation from a laboratory setup into a semi-industrial plant, as well as migration from a local control to an automatically controlled process involves complex activities. The objective of these activities is to identify the functions and sub-functions that must be performed by the equipment configuration in order to satisfy the requirements for operating in safety conditions. That involves monitoring and acquisi-

tion of the process parameters, complex parameter control, as well as specific calculation for each control loop from the plant, according to the process requirements (temperatures, flows, pressures). The system developed and described in the present paper is based on a SCADA (Supervisory Control and Data Acquisition) platform which contains: network communication, PLCs (Programmable Logic Controller) and HMI (Human Machine Interface). The control of process parameters has been done using the functions from the embedded programmable code of the PLC's which allows to have independent or combinations of PID components: proportional/integrator/derivative. Choosing the right control scheme and tuning PID loops corrects behavior of the system, improves the performance and leads to a safe operation of the LPCE process.

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P3.192 Comparative study of the neutronic performances of EU-WCLL and China-WCCB breeding blanket concepts

On the end of 2017, in the framework of EUROfusion R&D activities, a close collaboration between EU and China has started aiming at elaborating joint strategies for the development of the Water Cooled Lithium Lead (WCLL) and the Water Cooled Ceramic Breeder (WCCB) Breeding Blanket (BB) concepts. In this framework, an intense research campaign has been carried out at the University of Palermo, in close cooperation with ENEA Brasimone and ENEA Frascati, in order to compare the neutronics performances of the WCLL and WCCB BBs under irradiation in EU-DEMO.
To this end, three-dimensional nuclear analyses have been performed with MCNP5 v. 1.6 Monte Carlo code. A semi-heterogeneous MCNP model of the WCLL BB in a "single module segment" lay-out has been used and an analogous semi-heterogeneous MCNP model of the WCCB BB has been set up and adapted to the reference EU-DEMO model. The nuclear responses of both the BB concepts have been assessed focussing the attention mainly on global quantities as nuclear power deposition and Tritium Breeding Ratio as well as on the shielding performances. The obtained results provided significant information on the different nuclear features of the two concepts, useful for the optimisation of both WCLL and WCCB BBs design. The outcomes of this comparative study are herein presented and critically discussed.

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P3.196 Investigation of Fe-W interfaces in graded Fe/W composites for the first wall of DEMO

For the European demonstration power plant, four types of breeding blankets are under consideration. All designs agree in the basic materials selection, that is Eurofer used as structural material and tungsten used as armour material. Detailed thermo-mechanical finite element analyses show that a direct joint of these materials will not last over the anticipated lifetime of the blankets due to thermally induced macro stresses and strains.

An approach to cope with the thermal mismatch of W and steel is to implement a graded steel/W or Fe/W layer in the joint that re-distributes macro stresses. A novel, two-stages process has shown promising results in fabricating graded Fe/W composite materials. It includes, firstly, energetic mixing of Fe and W powders of several Fe/W ratios using a planetary mill. In a second step, Fe/W powders of different ratios are stacked and consolidated by mechanical pressure and electric heating within milliseconds. The process is referred to as Electro Discharge Sintering (EDS). Very short duration at elevated temperatures prevents the precipitation of intermetallic phases and, hence, allows assigning well-tailored properties to the graded composite. Analyses have shown that Young's modulus, strength and thermal expansion of Fe and W converge well across the graded layer whereas thermal conductivity remains on the Fe-level.

The current contribution focusses on Fe-W interfaces of the composites. Fe/W powders with 25, 50 and 75 vol.% W were mixed for 2, 4 and 8 hours and consolidated with identical EDS parameters (80 kN uniaxial pressure, 400 kA current). Based on a microstructural analysis, the impact of energetic mixing on Fe-W interfaces, and on physical and mechanical parameters is discussed. For selected samples, results of micro cantilever bending experiments that are carried out in a scanning electron microscope are presented. In-situ micro mechanical testing allows relating interface qualities to microstructures and mixing parameters.

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P3.198 Quantitative analyses of interfacial strain by diffraction contrast TEM imaging in ODS steel

Oxide dispersion strengthened (ODS) steel is one of the most promising candidate structural materials for fusion nuclear systems. It is widely recognized that to design and to control macroscopic materials properties of ODS steel successfully, a fundamental understanding of the atomic-scale structure and chemistry of oxide/matrix interfaces is necessary, owing to the fact that oxide/matrix interfaces plays a key role in determining the macroscopic properties. Lack of a statistical quantitative measurement method has been a major obstacle to analyze the complex strain field at the nano-scale particle/matrix interface.
In this study, an innovative diffraction contrast imaging method derived from Ashby-Brown contrast is presented. Fast and statistical quantitative measurement is achieved on the interfacial misfit strain around nano-sized semi-coherent Y2Ti2O7 particles in austenitic oxide dispersion strengthened steel, where nano-sized oxide particles were formed. A much lower average lattice misfit strain of 1.6%, rather than 9.45% expected from lattices of matrix and precipitate crystals, was revealed by this method, and confirmed with HRTEM. It illustrated that the nano-scale interfacial strain could be measured by measuring the inter-spacing between the so-called “no-contrast” lines, and the misfit dislocations around a large number of particles could be revealed by a single bright field TEM image. A correlation between nano-scale interfacial strain and the measurable TEM fringes was set up.
The mechanism of the measurement lies on the symmetry of the strain field at specific orientations, which induces “no-contrast” lines on TEM images. This method is especially suitable for measuring misfit strain around semi-coherent particles that are smaller than 10 nm. This study is expected to bridge the gap between nano-scale interfacial structure and mechanical properties of materials that are strengthened by semi-coherent nano-particles.

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P3.200 Preliminary Analysis on A Maintainable Test Cell Concept for IFMIF-DONES

The IFMIF-DONES (International Fusion Material Irradiation Facility- DEMO Oriented NEutron Source) is planned to deliver an intensive neutron source (5×10^16 n/s) for irradiation experiments that are confined and shielded by the Test Cell (TC). During the operation of the facility, unexpected degradation (by irradiation or corrosion) or damage (by handling etc.) of the TC leak tight liner, surrounding biological shielding walls, and active water cooling systems would need intensive repair work in an activated environment. Further technical difficulties will have to be confronted when the liner is tightly attached to the permanent massive concrete shielding walls with cooling water systems embedded inside.
A maintainable TC concept, which is targeting replacing TC key components in case of damages, is proposed in this paper. With this configuration, the TC liner and the primary shielding components with high failure possibilities are decoupled from the permanent construction structures, and are expected to be replaced in case of damage. Mechanical solution of this configuration is described

in this paper. Technical feasibilities of this configuration are preliminarily analyzed for the aspects of independent liner enclosure structure, interfaces with other IFMIF-DONES key systems, maintenance scenarios using remote handling systems, and etc.

Acknowledgments
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P3.203 Mechanical properties and microstructure of W/CuY and W/CuCrZr composites produced by hot isostatic pressing

It has been paid a great attention to the production of Tungsten/Copper (W/Cu) composites, as they appear promising materials to form part of the cooling system of the divertor of the future fusion reactors. However, further assessments of the microstructure and mechanical characteristics of these composites are required for the designs of the divertor. In this study, the mechanical behavior and the microstructure of W-15wt%CuY and W-15wt%CuCrZr have been investigated in compression from room temperature up to 500 °C at initial strain rate of 10-4 s-1. Both materials have been produced by mixing W powder and CuY and CuCrZr pre-alloyed powders followed by degassing and sintering hot isostatic pressing (HIP). Fully dense composites were obtained showing homogeneous distribution of pre-alloyed CuY and CuCrZr phase particles embedded in W phase. The yield strength for W/15 wt%CuY decreased with temperature from 620 to 360 MPa, while W/15 wt%CuCrZr exhibited a much less pronounced yield strength reduction in the same temperature range, i.e. from 400 to 320 MPa. The deformed microstructure at low (250 °C) and high temperature (500 °C) was characterized in the light of electron backscatter diffraction (EBSD) measurements and nanoindentation mapping. The correlation between the plastic deformation, grain structure and the orientation relationships in the composites has been revealed.

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P3.205 Physical design of GDT-based fusion neutron source with 2-3 folds increase of Q

Gas Dynamic Trap (GDT) is very attractive as a kind of fusion neutron source for testing fusion materials and components as well as driving fusion-fission hybrid reactor due to its linear and compact structure, low physics and technology requirement, relatively low cost and tritium consumption. These years, the conceptual designs of GDT-based neuron source for above two purposes, named FDS-GDT, have been proposed as a candidate of fusion neutron source by Institute of Nuclear Energy Safety Technology (INEST), Chinese Academy of Sciences • FDS Team in China. However, the fusion energy gains (Q) in current designs are still lower than 0.05.
In order to improve the Q and reduce the technology requirements of magnet and neutral beam injection (NBI) for GDT based fusion neutron source, a new method was proposed that using high field neutral beam injection (HFNBI) to replace the neutral beams obliquely injected at middle plane of GDT where the field is minimal. This method will benefit for confining higher density of fast ions at the turning point in the zone with a higher magnetic field, as well as getting higher mirror ratio by reducing mid-plane field rather than increasing the mirror field. The preliminary analysis with system code SYSCODE indicates that the fusion energy gain can be improved 2-3 folds.
With HFNBI method, two designs of GDT-based fusion neutron source (FDS-GDT) were updated, one is for testing fusion materials and components with 2.57MW of fusion power, 2MW/m2 of neutron flux density and 0.13 of Q, and the other is for driving fusion-fission hybrid reactor with 14.72

MW of fusion power and 0.16 of Q. Further detailed simulation will be carried out to optimize and verify the designs by 3D Monte-Carlo simulation code MCFIT and bounce-averaged simulation code DOL developed by Budker Institute of Nuclear Physics (BINP), Russia.

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P3.207 Microstructural comparison of Oxide-Dispersion Strengthened Ferritic Steels produced by HIP and Spark Plasma Sintering

The high mechanical strength of ODS FS, and their resistance to creep and neutron radiation damage up to 750 ºC are attributed to extremely fine microstructures with high density of very stable nanometric precipitates, generally Y-Ti-O oxides. The STARS route (Surface Treatment of gas Atomized powder followed by Reactive Synthesis) proposed by Ceit avoids mechanical alloying to introduce yttrium in atomized prealloyed powders. Powders containing yttrium are atomized and oxidized to grow a thin metastable Cr-, Fe-rich oxide layer on the surface of particles. During HIP consolidation at high temperature, oxides at Prior Particle Boundaries (PPBs) dissociate, oxygen diffuses towards titanium and yttrium, and Y-Ti-O nanoparticles precipitate.
Once the feasibility of the STARS route has been validated, two strategies are proposed in this work to enhance precipitation of nanoparticles and improve the mechanical strength. Firstly, consolidation by Spark Plasma Sintering (SPS) is evaluated to produce finer microstructures compared to HIP consolidation. SPS parameters like temperature, heating rate, pressure and time of exposure under pressure were explored using spherical powders, and dense samples (>99,5 %RD) were obtained.
Secondly, a profuse precipitation of nanoparticles is achieved through a combination of HIP consolidation at low temperature (700-900 ºC) followed by hot deformation under the presence of metastable oxides, to increase the density of dislocations. These dislocations are preferential nucleation sites for the precipitation of nanometric Y-Ti-O oxides during final heat treatment at high temperature (>1200 ºC). Deformation dilatometry and plane strain compression tests were performed to simulate hot deformation under different hot rolling schedules. Total deformation, number of passes, time between passes or initial and final rolling temperatures are some of the parameters explored.
Microstructural characterization by Scanning Electron Microscopy (SEM) and Transmission Electron Microscopy (TEM) of consolidated materials is presented and correlated with mechanical behavior.

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P3.209 Characterization and thermomechanical assessment of a SiC-sandwich material for Flow Channel Inserts in DCLL blankets

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Flow Channel Inserts (FCIs) are key elements in the Dual Coolant Lead Lihitum (DCLL) blanket since they decouple electrically the flowing PbLi and the blanket steel structure, minimizing the MHD pressure drop. Furthermore, in the high-temperature version of the DCLL (where the PbLi may reach temperatures up to 700 ℃), FCIs also protect the steel structure from the hot liquid metal. The material for FCIs should present adequately low thermal and electrical conductivities, together with good stability against hot PbLi and enough mechanical integrity to withstand mechanical stresses derived from high thermal gradients during operation. SiC-based materials are candidates for high-temperature FCIs due to their high stability at high temperatures; specifically, a dense-porous SiC sandwich material with sufficiently low conductivities but offering reliable protection

against PbLi corrosion and infiltration is an attractive option.
In the present work, a dense-porous SiC sandwich material is fabricated and characterized. The porous SiC core is manufactured from SiC powder by two different techniques, uniaxial pressing and gel-casting (the latter being industrially up-scalable); the porosity is introduced using graphite spherical powder as sacrificial template. After production of the porous SiC core, a dense SiC coating of ⊠200 ⊠m thickness is deposited by CVD. The properties of the produced material in terms of thermal and electrical conductivities (the last one including the effect of ionizing irradiation), flexural strength and stability against both static and flowing PbLi (addressing the possible influence of a magnetic field in the corrosion phenomena) are presented. In order to assess the suitability of the material for FCIs in a high-temperature DCLL, stress analysis and heat transfer simulations were also performed, taking into account the situation in the blanket (including possible MHD effects) and the properties of the material produced. The main results of these simulations are presented.

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P3.211 Electrochemical techniques as innovative tools for fabricating divertor and blanket components in fusion technology

Electrochemical techniques such as electroplating of metals, electrochemical machining (ECM), electroforming, anodizing and electropolishing of metal surfaces have been established successfully in a variety of industrial processes. A wide range of applications are available such as the electrodeposition of decorative metal coatings on plastics and metals, corrosion protection of mass products like steel sheets, and the electrodeposition of high electro-conductive coatings in electrical industry. The main advantages of these electrochemical processes are good controllability of coating thicknesses, their cost-effectiveness and their good scalability from scientific settings to industrial scale.
However, for applications in fusion technology material development needs to meet unique and specific restrictions or requirements, e.g., low activation, high heat-flux properties, corrosion resistance to liquid metal alloys, e.g., Pb-15.7Li, T-permeation reduction and suppression of MHD effects. Therefore, uncommon materials like tungsten, CuCrZr, RAFM-steels have to be processed for which available standard electrochemical processes show various limitations at the moment.
To overcome these limitations, electrochemical methods were introduced to divertor and blanket component manufacturing that led to promising electrochemical techniques contributing their inherent advantages to a variety of applications. An example for such developed processes is the ECX process, which is based on the electrodeposition of aluminum from ionic liquids, to fabricate corrosion barriers on RAFM steels that are resistant against corrosion in flowing Pb-15.7Li. Others are the electrodeposition of metals (Cu, Pd, Ni) on tungsten and CuCrZr substrates for joining applications and the surface microstructuring of tungsten surfaces by ECM in first wall applications.
This paper provides an overview on these processes (ECX, electrodeposition on W and CuCrZr, ECM) and describes the achieved benefits of using electrochemical material processing technologies for applications in fusion technology. Additionally, potentials for transferring these processes onto industrial scale will be highlighted and provide implications for future directions of scientific research and development.

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P3.213 Design of the Main Building and Plant Systems of the IFMIF-DONES facility

In the framework of the EU fusion roadmap implementing activities, an accelerator-based Li(d,n) neutron source called DONES (Demo-Oriented early NEutron Source) is being designed within the EUROfusion workpackage WPENS as an essential irradiation facility for testing candidate materials for DEMO reactor and future fusion power plants. The objective of this workpackage is to be ready for IFMIF-DONES construction as soon as 2020.

Recently a joint Spain-Croatia proposal to host DONES in Europe (Granada, Spain) has been made. The facility configuration is based on a number of systems grouped by areas with common technologies (the Accelerator Systems, Tests Systems and Lithium Systems). Besides them, the Site, Buildings & Plant Systems need to be designed in order to provide the necessary services. Also, the Main and the Auxiliary Buildings to host all the DONES Structure, Systems and components are to be designed.

In this paper, the engineering works done for the Site, Buildings & Plant Systems as well as the Main Building design activities will be presented. Within these engineering and design works the adaptation from a generic to the Granada specific site will be included, as well as the consolidation of the engineering input data that will allow a detailed engineering and the production of the technical specifications to be issued to the subcontractors. Also, the design criteria as well as the design basis for the facility Structures, Systems and components is included.

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P3.216 Commissioning of The Materials Detritiation Facility at Culham Science Centre

The Materials Detritiation Facility (MDF) has been designed to thermally treat solid non-combustible radioactive waste produced during operations of the Joint European Torus (JET) that is classified as Intermediate Level Waste (ILW) in the UK due to its tritium inventory (>12kBq/g). The wastes primarily comprise Inconel, steels, aluminium alloys, copper and carbon-based composites. These materials will be thermally treated in a retort furnace at temperatures up to 1000℃ under a flowing air atmosphere to reduce their tritium inventory sufficiently to allow disposal at a lower waste category. The tritiated exhaust gases from the furnace will be processed through a bubbler system, where released tritium will be trapped in water.
Commissioning of the facility has been divided into two constituent parts: inactive and active. The inactive commissioning is to verify that all components and safety systems of the facility are installed, functioning and operating within their design limits. Several inactive trials of the thermal treatment process will be performed on each material type.
During the active commissioning, small amounts of tritium-contaminated material will be introduced into the process and used for limited active trials. The tritium inventory in this material has been selected to be As Low As Reasonably Practicable (ALARP), ensuring that activity levels are sufficient to fully test the process design and control instrumentation but pose minimal risk to operators. The active trials will include four thermal heating cycles on carbon-based and Inconel materials incorporating increasing levels of tritium (ca. 1MBq, 3GBq, 20GBq and 26GBq respectively). Materials used in the active trials will be sampled and analysed before and after thermal treatment to verify the performance of the process and confirm that the bulk of the tritium has been removed from the solid matrix by the heating process and has been recovered into the water within the bubbler system.

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P3.217 Fracture toughness characterization of Eurofer97 steels in EUROfusion using miniature multi-notch bend bar specimens

Eurofer97 is one of the leading candidates of reduced activation ferritic martensitic (RAFM) steels for first wall structural materials of early demonstration fusion power plants. During fusion plant operation, intense neutron irradiation damage on first wall materials can cause significant irradiation embrittlement and greatly reduce the fracture toughness of RAFM steels. Therefore, it is critical to characterize the fracture toughness of RAFM steels with high fidelity. This is especially true for testing neutron irradiated specimens which are usually much smaller than standard size specimens due to limited volume of irradiation facilities and low radioactive dose advantage of small size specimens after irradiation. Under the framework of EUROfusion, we are in the process of characterizing fracture toughness properties and irradiation embrittlement behavior of ten Eurofer97 steels with different compositions and thermomechanical treatments. We have developed fracture toughness testing technique using pre-cracked miniature multi-notch bend bar specimens (M4CVN) with a dimension of 45mm (length) x 3.3mm (width) x 1.65mm (thickness) based on the ASTM E1921 Master Curve method. Dedicated experimental setup has been designed and fabricated for testing the M4CVN specimens for both regular laboratories and hot cell facilities at Oak Ridge National Laboratory (ORNL). Fracture toughness results, along with tensile, microhardness, and microstructure results, will be discussed to elucidate the link between materials microstructure and fracture toughness properties for unirradiated conditions. Neutron irradiation for ten Eurofer97 steels at 300℃ for 2.5 dpa (displacement per atom) is currently underway at High Flux Isotope Reactor of ORNL and depending on the test progress, post irradiation examination results may also be presented to study the irradiation embrittlement behavior of Eurofer97 steels.

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P3.222 Impact of external costs on the penetration of fusion power plants in the future global electricity system

Externalities are defined as a cost that arises when the social or economic activities of one group of persons have an impact on another group and when that impact is not fully accounted, or compensated for, by the first group (ExternE project). External costs are not usually considered in the total cost of electricity causing market failures. To fairly compare the different electricity generation technologies in terms of costs and benefits, externalities should be included when estimating the total costs per kWh produced. In this work, a comparison among the external costs of the electricity generation technologies, including fusion, is presented. Then, those costs have been introduced into the EUROfusion Times Model (ETM) to analyse their impact on the global electricity system evolution in the long term with special focus on the penetration of fusion technologies. ETM is a global optimization energy model developed within the framework of the Socio Economic Studies project (SES) in EUROfusion, which provides the optimum energy system at minimum cost and maximum social welfare and sustainability.
In a first step, four scenarios have been formulated and represented by ETM. A Reference scenario, and a Policy scenario which represent a highly electrified global system, both with and without external costs. Results show that when external costs are included in the total costs of electricity, fusion technologies become more competitive as their associated externalities are in the range of some renewable and nuclear fission technologies in contrast to those from the fossil fuel technologies.

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P3.226 How fusion power can contribute to a fully decarbonized European power mix after 2050.

In the second half of this century, the European energy mix will be very likely completely decarbonized. Two main options are available to generate carbon free electricity: either to rely on renew-

able energy sources or to further differentiate the energy mix by including nuclear power.
In the former case a large storage capacity and/or back-up dispatchable generation are required to compensate for the intermittent electricity production from variable renewable sources. The size of the necessary storage system and back-up power plants can be reduced if a base-load carbon free power technology is available. Indeed, the affordability of a power mix mainly based on solar power (South Europe) is likely to increase if coupled with a large base-load generation with power swing that systematically reduces the day time power output. Conversely, in case of a power mix mainly based on wind power (North Europe), power swing is not necessary.
These options are studied with the COMESE code that assesses the hourly balance between load and generation through a simplified dispatch model and the system costs through a stochastic economic analysis.
We assume a complete phase out of fission power plants by 2080 in Europe, so that nuclear electricity is provided by fusion power plants only, operating in continuous or power- swing regimes. The probability distribution of the cost of electricity resulting from the stochastic economic analysis of each scenario gives key indications to make fusion a cost- effective ingredient of a future European decarbonized power mix.

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P3.231 Neutronics study for DTT tokamak building

The Divertor Tokamak Test (DTT) facility is a project proposed by the Italian Consortium aimed to test the physics and technology of various alternative divertor concepts in order to design a heat and power exhaust system able to withstand the large loads expected in the divertor of a DEMO fusion power plant.
Even though DTT is a machine operating without tritium, a significant 2.5 MeV neutron yield by deuterium-deuterium (DD) reactions is foreseen. During high performance scenarios a maximum DD neutron yield rate of 1.5⊠1017 n/s is expected and according to the present operation plan, 3.7x1022 neutron will be produced during its lifetime with an annual budget ranging between 9x 1018n and 1.53x1021 n. Furthermore, due to triton burn-up a significant 14 MeV neutron production is also expected (1-2% of the total yield).
Therefore neutronics and activation analyses are fundamental for the design of the machine as well as for licensing, maintenance, decommissioning and waste management.
This work is devoted to preliminary three-dimensional shielding study optimize the DTT building for licensing purposes. Neutron and gamma transport simulations were carried-out with MCNP5 Monte Carlo coupled with FENDL nuclear data libraries code and the ADVANTG tool has been used for the implementation of specific variance reduction techniques. 360° model of DTT machine and building have been developed to assess the spatial distributions of the neutron and gamma fluxes, spectra and effective doses inside and outside the tokamak hall. The results of the shielding study and the impact of building design on radiological protection issues are presented and discussed.

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Present Progresses and Activities on the Chinese Fusion Engineering Test Reactor

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The Chinese Fusion Engineering Testing Reactor (CFETR) is the next device for the Chinese magnetic confinement fusion (MCF) program which aims to bridge the gaps between the fusion experiment ITER and the fusion power plant. CFETR detail engineering design and R&D project have been

approved by Chinese government. The activities have been started in the end of last year. CFETR new design focus on the high magnetic field by using high performance Nb3Sn wire for TF and 2212 HTc CICC for CS, and large size with major radio 7m. Steady-state operation and tritium self-sustainment will be the two key issues for the first phase with a modest fusion power up to 200 MW. The staged operation for late phases will explore for DEMO validation with a fusion power over 1 GW and Q over 20. Operational scenarios with L-mode, hybrid H-mode and steady state advanced H-mode physics will introduced in this presentation together with R&D activities for H&CD, diagnostic, VV, divertor, superconducting magnets, T-plant and related technology, material, remote handling, physical validation on EAST tokamak aiming high performance steady state operation, and future developing plan.

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European Materials Development: Results and Perspectives

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This paper reviews the material strategy of the EU fusion roadmap and the recent progress of activities within the EUROfusion materials research program. It highlights, both, the characterization and validation of in-vessel components baseline materials, i.e. EUROFER97, CuCrZr and tungsten as well as the development and characterization of advanced structural and high heat flux materials for DEMO and beyond. In support of engineering design activities the primary focus is on compilation of data and the supply and release of material property handbooks, material assessment reports complemented with the development of design criteria and material design limits appropriate for DEMO thermal, mechanical and environmental conditions.

Data are presented and discussed with respect to DEMO operating specifications for selected subtopics, which include (a) advanced steels optimized towards low and high temperature extension of the current operational window, (b) heat sink materials, i.e. copper based alloys and composites and (c) plasma facing materials, i.e. tungsten based composites. Based on the discussion and conclusions drawn, perspectives for the required materials performance and future research and validation steps will be given.

A first glance on these perspectives, the most far-reaching progress by now in the EUROfusion materials program and the first step of a necessary and more extensive qualification program, is provided by the launch of nine neutron irradiation campaigns within the last two years. Baseline structural and high heat flux materials are irradiated up to medium neutron dose levels for continuously filling gaps in the materials property handbook and advanced material options at lower neutron fluence for screening, down-selection and increase of fundamental knowledge of n-damage. The results will guide the future materials research and validation program as well as design options for blanket and divertor components.

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Towards the EU fusion-oriented neutron source: the Preliminary Engineering Design of IFMIF-DONES

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The need of a high-intensity, 14 MeV-peaked neutron source for the qualification of materials under fusion-relevant conditions has been recognized in the European (EU) fusion programme as an essential step towards the design and licensing of DEMO and future commercial fusion power plants.

This need has pushed the EU to support the development of a Li(d,nx) neutron source called IFMIF-DONES (International Fusion Materials Irradiation Facility-DEMO Oriented Neutron Source) based on and taking advantage of the results obtained in the IFMIF/EVEDA (Engineering Validation and Engineering Design Activities) project conducted in the framework of the bilateral EU-Japan Broader Approach Agreement.
The design activities and the supporting R&D work of the DONES facility are presently being carried out in the framework of the Work Package Early Neutron Source (WPENS) of the EUROfusion consortium in close collaboration with Fusion for Energy agency, with the main goal of consolidating the underlying technology and developing a sound design basis in order to be ready for IFMIF-DONES construction at the early beginning of the next decade.
In this paper, the main engineering advances achieved during the first three years of the WPENS project and included in the recently released IFMIF-DONES Preliminary Engineering Design Report as an important milestone of the project are presented, focusing in particular on the main design evolutions from the previous phases and on the critical aspects to be further developed in the near future.
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P4.195 Stress-strain characterization of abs polymer at cryogenic temperatures

Plasma Laboratory for Fusion Energy and Applications at Instituto Tecnológico de Costa Rica (ITCR) plans the design of SCR-2; a Quasi-Toroidally Symmetric (QAS) two-field period modular Stellarator, aspect ratio ~5 formed by 24 modular magnetic coils. SCR-2 coils design is based on ESTELL QAS configuration (project cancelled) [1], supplied by Max Planck Institute for Plasma Physics, Greifswald, Germany.

Among SCR-2 design aspects, superconductive coils (SCC) and an additively manufactured (AM) coils structure stand out. These present engineering challenges, since high precision complex geometries [2][3] and cryogenic environment conditions must be complied at reasonable cost. Besides, AM research for magnetic confinement device construction is not yet a vast study field. Hence, this technique must be regarded for preliminary studies.

First, 77 K stress-strain (SS) tests on AM ABS specimens are carried out at ITCR, considering several manufacturing parameters. These parameters´ influence on the results obtained is analyzed using Design of Experiments methodology. In order to perform SS tests, a LN2 vessel adaptation for an universal testing machine was built. Tests description, results and SCR-2 concept models will be presented. Results from these experiments will be reckoned with so that FEM simulations of a prelusive design for the 3D printed modular coil structure could be performed.

[1] M. Drevlak et al., "ESTELL: A Quasi - Toroidally Symmetric Stellarator", Contrib. to Plasma Phys., vol. 53, no. 6, pp. 459 – 468, 2013.
[2] V.I. Vargas et al, Implementation of Stellarator of Costa Rica 1 SCR 1, in: 26th IEEE Symposium on Fusion Engineering (SOFE), 31 May – 4 June 2015, Austin, TX (USA), IEEE Conference Publications, 2016, pp. 1–6, ISBN: 978–1-4799–8264-6.
[3] V. Queral, "Prospects for Stellarators Based on Additive Manufacturing: Coil Frame Accuracy and Vacuum Vessels", IEEE Transactions on Plasma Science, issue 99, pp. 1-7, 2018.

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P4.121 Testing of a high temperature radiatively cooled Li/Ta heat

pipe in Magnum-PSI

In 1998 Makhankov [1] described the concept of modular exchangeable plasma facing components (PFCs) based on liquid metal heat pipes which are radiatively cooled. Here we present results from recent experiments with a lithium filled tubular heat pipe owned by Sandia National Laboratories. The tantalum envelope (~20mm diameter by ~200mm long) was heated on its side wall using a hydrogen plasma beam in Magnum-PSI. A single continuous pulse lasting ~2 hours was carried out with the active zone of the heat pipe operating at a temperature of ~1000⊠C for the whole time with no cooling other than the Planck radiation. Target tilting was used to vary the peak surface heat flux in the range 8-14MWm2. The tilting also increased the magnetic field component normal to the return flow of lithium via the sintered niobium wick to ~1T to allow evaluation of the impact of the magnetic forces. Near infra-red thermography from two orthogonal ports was used to diagnose the surface temperature distribution. Reference pulses run on a 2mm thermally isolated molybdenum plate provided a simple target for beam power characterisation.
The heating power was gradually increased until liquid lithium escaped through a crack in the heat pipe near the beam centre [2]. The lithium loss was insufficient to affect heat removal right up to the time when the beam was switched off. The impact of the lithium leak on the plasma was benign compared to that expected from leaks in helium or water cooled PFCs. Other benefits of radiatively cooled modular PFCs are also reviewed in this presentation. Future materials and technology are discussed in [2].

[1] A.Makhankov, Fusion Engineering and Design 42 (1998) 373–379
[2] R. Nygren, SOFT 2018 "Post-test examination of a Li-Ta heat pipe exposed to H plasma in Magnum-PSI"

Part-funded by RCUK Energy Programme grant EP/P012450/1.

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P4.128 Locally adapted heat transfer using variable height rib structures for Optimization of HCPB DEMO FW cooling channels

Several European First Wall (FW)/blanket concepts for DEMO are high-pressure (8 MPa) Helium-cooled systems. Typical radiative steady state loads delivered from the fusion plasma are predicted up to 0.45MW/m², but peak values could reach and excess 1MW/m². At such high heat fluxes, it is a major engineering challenge to dimension the cooling for moderate temperatures and moderate stresses, while limiting the coolant pumping power. Ribs promoting turbulence and secondary flows in the coolant have already been identified by experiments and Computational Fluid Dynamics as effective measure to increase to heat transfer coefficients by about factor two, with only a moderate penalty on increased pumping power

In a previous publication, a design point capable to sustain heat flux densities of 1MW/m² at an average shell temperature lower than 500 ℃ has been found by running thermal-mechanical simulations for a FW channel segment, with correlations to describe heat transfer and pressure drop. In this paper, we apply hybrid RANS/ LES methods at high Reynolds numbers to predict the detailed temperature distribution in the structures. The FW channel is a two-side heated (breeder side and plasma side), one-side rib-roughened FW channel with rounded corners. Local heat transfer tuning like increasing the heat transfer coefficient along the channel should economize the pressure drop. Therefore this paper deals with the investigation of upstream 60° V-shaped ribs elements with squared cross section with a rib-pitch-to-rib-height-ratio of 10 and a rib-height-to-hydraulic-diameter-ratio in the range of 0.072. The heat transfer coefficients for promising configurations and the pressure drops compared to smooth channels are identified. Strategies in order to reduce the mesh count and its subsequent impact on the accuracy of the solution are assed in order to bring down computing time down considerably to be able to calculate a complete first wall channel.

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P4.130 Characterisation of JET ILW divertor plasma facing surface materials by temperature-QMS analysis

Fusion fuel retention is a major issue for fusion devices. A set of divertor HFGC tile samples from first 2011-12 JET ITER-like wall (ILW-1) campaign were analysed in the research to complement the data obtained by other analysis methods [1].
Experiments were carried out using Hositrad MGT 6-300 Multi Gas Analyser with Thermal Desorption. The device operates with two Quadrupole Mass spectrometers (QMS) – high precision 1-6 amu detector (0.02 amu resolution) and 6-300 amu detector (0.1 amu resolution) with heating up to ~1470 K in vacuum (p ≈ 1•10^-6 mbar). Analysed samples were heated to 1273 K (~15 K•min-1 heating rate).
Investigation of plasma facing surface samples provides signals of relatively short (5-8 C) hydrocarbon CxHy fragments upon initial heating till ~600 K as evidence of thin organic films on the surface of divertor tile. Species of fusion fuel (D/D2/DT/T) were observed though release temperature and amount varies with samples. Initial T+ release peaks were observed at ~700 K, with secondary peaks at ~1000 K and ~1200 K.
Results were compared to tritium activity from full combustion experiments done previously and photo-stimulated luminescence (PLS) data [2] of the same samples. Though at present stage of the study obtained data cannot be fully quantified, amounts of species observed are corresponding to PLS results.
Overall, QMS analysis confirms the presence of fusion fuel components in the divertor material plasma facing surface similarly observed by other analysis methods.

[1] A.Widdowson et.al./ Nuclear Fusion 57 (2017) 086045
[2] Y.Hatano et.al./Journal of Nuclear Materials 463 (2015) 966–969

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

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P4.210 Fatigue behaviour of prospective fusion oxide dispersion strengthened CoCrFeNiMn high entropy alloy

Fatigue behaviour of the oxide dispersion strengthened CoCrFeNiMn high entropy alloy was characterised at the first time. The initial powder was prepared via mechanical alloying and the material densification was done by the SPS technique. A microstructure, as well as basic mechanical characteristics, were obtained from the tensile test. These data allowed to set parameters for fatigue experiments to be partially in the plastic regime. The fatigue data were obtained on the bars loaded in bending with R=0.1 loaded up to the final fracture or 107 cycles. The fracture surfaces were analysed and fracture initiation places (origins) were identified. Frequently larger grains with extensive twinning formation were such origins. The TEM observation was conducted on the foils extracted from the fracture surfaces to confirm expected damage mechanism observed on the fracture surfaces. The oxide dispersion strengthened HEA was compared with a non-strengthened variant of the same alloy. The shift to higher stress levels for the same lifetime was observed for the strengthened high entropy alloy.

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P4.095 Design and thermal coupling analysis of vpi mould for iter pf6 double pancake

The ITER PF6 coil is composed by 9 double pancake(DP).The vacuum pressure impregnation（VPI）technology is an important process for double pancake manufacturing.The double pancake winding is contained within the VPI mould consisting of vacuum chamber and hard mould.The vacuum chamber provide the vacuum and pressure conditions required by VPI process,The hard mould provide the heating and shape requirements of VPI process.Using structure mechanics and thermodynamic analysis with finite element to check and optimize the DP VPI mould model.Check and optimize the DP VPI mould through actual VPI process of the Dummy double pancake.The results show that DP VPI mould can meet the requirement of VPI process By now,5 double pancakes(Include Dummy DP) VPI has been successfully accomplished.

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P4.226 The monitoring and determination of Beryllium contamination at HELCZA facility

This paper presents application and optimization of the method for the quantification of beryllium dust particles in experimental complex HELCZA (High Energy Load CZech Assembly) by selected available analytical instruments mainly by portable fluorometry as a procedure ensuing from the NIOSH (National Institute for Occupational Safety and Health) method. The experimental complex HELCZA, a high heat flux test facility (HHFTF), is designed for cyclic heat loading for testing ITER plasma-facing components, for example, the first wall panels. These tests go along with the need to control the airborne particles of beryllium because the accurate knowledge of present beryllium in the workplace is crucial considering its high toxicity. With HELCZA being put into operation, the great attention is dedicated to safety procedures including to measure beryllium dust particles.

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P4.011 An innovative helicon plasma source for an alternative concept of DEMO negative ion beam injector

Neutral Beam Injectors (NBI) for DEMO-like reactors will need deuterium neutrals at a high energy (>0.8MeV) and a fair injector overall efficiency (>50%) for plasma heating and current drive. The neutralization efficiency of positive ions drops for energy higher than 100keV/nucleon and so NBIs based on negative ions are required. A conceptual design of injectors (so called Siphore at the IRFM in the CEA, Cadarache in France) expects to extract negative deuterium ions from a plasma 3m long and 15cm wide, and to photo-neutralize the accelerated D-.
Here, we review the progress at the Swiss Plasma Center of EPFL to develop an innovative helicon device (so called RAID: Resonant Antenna Ion Device). It consists of a resonant network plasma source at 13.56MHz (deliver power ≥10kW), connected to a cylindrical vacuum chamber (1.8m long, 0.4m diameter), surrounded by 6 Helmholtz coils providing a magnetic field (up to 800G on axis).
RAID has proven high reliability to propagate helicon wave in different gases (Ar, D2, H2) and pressures (0.1-3Pa). Indeed, the resonance properties of the antenna enables easy power matching with low input current. Spectroscopy measurements have shown a significant volume production of negative ions at the periphery of the column, explained by a strong radial gradient of electron temperature. Today, RAID is equipped with a full set of diagnostics (optical emission spectroscopy, Langmuir probes, cavity ring down spectroscopy and photodiodes) and it will be complemented in the coming months by a 3-axis magnetic probe for a full characterization of helicon wave field.

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P4.012 SPIDER BEAM SOURCE READY FOR OPERATION

In ITER, each Heating Neutral Beam injector (HNB) will deliver about 16.5 MW heating power by accelerating a negative ion beam up to the energy of 1 MeV for a duration of 1 hour.
To this purpose, a large RF-driven plasma source is required to generate a 40A D- or 46A H- ion current, with low electron/ion ratio (<1) and high uniformity over the extraction area (800 mm x 1600 mm). SPIDER experiment is meant to demonstrate the performance of a full scale prototype ion source, complying with the challenging requirements of the ITER HNBs.
The SPIDER Beam Source (BS) was delivered to the Neutral Beam Test Facility (NBTF) site in Padova (Italy) in October 2017 by the European Domestic Agency, after about five years procurement phase. A huge effort was devoted during the procurement for quality controls and testing of the BS at the supplier's workshop. However several activities were carried out on NBTF site for verification of interfaces, solution of still open issues, as well as final tests before installation inside the Vacuum Vessel.
The NBTF Team undertook the BS site acceptance tests including: pressure and leak tests of the hydraulic circuits; electrical tests; measurement of magnetic field profiles; functionality tests of diagnostics installed on the beam source; checks of grids alignment by means of laser tracker.
Accurate positioning of the BS inside the vacuum vessel was performed on-site and various service lines were connected in a very limited space. Several improvements were undertaken in order to guarantee reliability and reduce the risks during the first commissioning and experimental campaign in 2018.
After installation, the integrated commissioning phase was initiated, powering the RF and high voltage circuits. This phase will be followed by the first operation in vacuum with negative ion production and low voltage extraction and acceleration.

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P4.035 Command and control system for STAR X-Ray source

The Southern Europe Thomson Backscattering Source for Applied Research (STAR) is a compact hard X-ray source designed by INFN for advanced applied materials-science research and founded by Progetto
MaTeRiA. MaTeRiA has been realized thanks to a partnership between the University of Calabria and CNISM (Consorzio Nazionale Interuniversitario per le Scienze fisiche della Materia).
The machine has been installed at the University of Calabria (UNICAL); the X-ray beam is produced using the Thomson Back-scattering process, where relativistic electron bunches, accelerated by a 60 MeV Linear Accelerator (LINAC), interact with very short photon pulses, produced a by a 30 mJ@5 ps IR (Infrared) Pulsed Laser.
The EPICS-based command/control (ComCon) system of the STAR machine has been designed and currently is under commissioning, using technologies at the state of arts and making extensive use of COTS hardware and software assuring full control and supervision of the whole machine and all relevant subsystems: Beam Diagnostics, Magnets Power Supply, RF devices, Laser systems, Vacuum subsystem, Machine Protection System, Personnel Protection System interface and timing system.
A growth potential has been considered in the design phase, in order to take into account any future expansions of the control system during the lifecycle of the facility without impact on the performance.

Scalability of systems is crucial, for the moment the current design allows to generate tunable collimated monochromatic X-rays in the range between 10 and 200 keV, in the future the main idea will be to obtain a machine to generate higher energy X-ray beams, towards the MeV. The control system from both SW and HW points of view has to accommodate this need as well as allowing to operate the machine for a long lifecycle.

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P4.002 Characterization of the input beam to RFQ of the Linear IFMIF Prototype Accelerator (LIPAc)

The Linear IFMIF (International Fusion Materials Irradiation Facility) Prototype Accelerator (LIPAc) is a key activity to demonstrate the validity of the low energy section of an IFMIF deuteron accelerator up to 9 MeV with a beam current of 125 mA in CW. For the successful acceleration of the high power beam, the input beam to the 5 MeV LIPAc RFQ should be fully characterized and controlled to achieve the good performance and stability. H+ beam with half energy (50 keV) and half current (70 mA) but same perveance with respect to the nominal D+ beam for the LIPAc injector is used at the initial phase of the beam injection into the RFQ. The objective is to characterize the performance of the RFQ, prove the ability of the MEBT to match the beam to the nominal SRF Linac beam parameters and test the beam diagnostics that will be used later on to characterize the beam characteristics in CW with less beam power and without the risk of the activation induced by a D+ beam. Furthermore, at the very beginning of the commissioning, it is requested to inject the pulsed beam with the current of 10 mA or less for check-out of each component. This paper describes the latest results obtained through the efforts to characterize the pulsed H+ beam with 50 keV energy at various beam current, 10 − 70 mA. After the maintenance of the accelerator column of the ECR ion source performed in 2017, the quality of the extracted beam was much improved than before, and the normalized rms emittance is less than 0.15 pi.mm.mrad which is about half of the acceptance value. The plan to achieve the stable CW beam with the same quality based on the present experimental results is also presented.

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P4.083 Concept design of high power quench protection circuit on China Fusion Engineering Test Reactor

The Quench Protection Circuit (QPC) can provide a fast and reliable protection for superconducting magnet. According to the conceptual design of China Fusion Engineering Test Reactor (CFETR), a high power QPC with mechanical-static hybrid switch is proposed, which is designed for the superconducting Toroidal Field (TF) magnets. The rated currents and maximum reapplied interruption voltages for QPC are 70 kA and 15 kV. The QPC is mainly composed of a innovative mechanical main switch with three levels contacts for conducting the continuous current, a Solid-State Switch (SSS) in parallel for current transferring, a high power resistor for dumping energy stored in the coils, and a pyrobreaker for providing backup. The design principles and preliminary results of each part are given respectively.

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P4.004 Characterization of JET neutron field for radiation studies in DD DT and TT plasmas

A new Deuterium-Tritium campaign (DTE2) is planned at JET in 2019/20, with a proposed 14 MeV neutron budget nearly an order of magnitude higher than any previous DT campaigns. With this proposed budget, the achievable neutron fluence on the first wall of JET will be up to about 10E20 n/m2, comparable to that occurring in ITER at the end of life in the rear part of the port plug, where several diagnostic components are located. The DT operation of JET will be preceded by a campaign with a tritium plasma.

The purpose of the present work is to characterize the neutron and gamma ray field inside the JET device. An analysis of the neutron/gamma ray flux, energy spectrum and dose rate levels is performed at selected irradiation locations, such as the neutron activation irradiation ends, the long term irradiation stations located inside the vessel and inside a circular horizontal port, where samples would be exposed to the maximum neutron flux or fluence. The calculations at the selected irradiation positions are performed for the case of a DD, DT and TT plasma and compared.

Calculations are performed with the use of the MCNP code in combination with the variance reduction code ADVANTG, enabling superior statistical results to previous calculations and allowing a good comparison of the spectra and fluence between individual locations for all three types of plasma.

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P4.006 Design review for the Italian Divertor Tokamak Test facility

The original DTT (Divertor Tokamak Test facility) proposal presented in 2015 [1] in the challenging area of plasma exhaust has been described in detail in [2] with a critical review of several aspects.
Afterwards, according to the conclusions of the DTT Workshop held at Frascati in June 2017, various points have been examined to improve the proposal in the design review phase.
One issue is the flexibility of the machine, so as to be able to incorporate the best candidate divertor concept (e.g. conventional, snowflake, super-X, double null, liquid limiter). In particular, up-down symmetry has been required so as to properly test double null configurations. Another important point is the additional power coupled to the plasma needed to guarantee significant results in view of DEMO.
Here we present the engineering aspects of the Italian DTT design review, which led to an up-down symmetric tokamak with a major radius of 2.08 m, a minor radius of 0.65 m, a toroidal field of 6 T, a plasma current of 5.5 MA, and a pulse duration of 90-100 s.
In particular we illustrate the rationale for the design choices, focusing on the main differences with respect to [1-2], namely for:
• magnet system;
• scenarios;
• vacuum vessel;
• in-vessel components;
• thermal shield and the neutron shield;
• additional heating system.

[1] DTT Divertor Tokamak Test facility, Project Proposal, ENEA, 2015, 2016, ISBN:978-88-8286-318-0, http://fsn-fusphy.frascati.enea.it/DTT/downloads/Report/DTT_ProjectProposal_July2015.pdf,
[2] Special Section of Fusion Engineering and Design, Vol. 122, 2017, pp. 253-294 and e1-e25

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P4.009 Technological challenges for the design of the RFX-mod2 experiment

An upgrade of the RFX-mod experiment is presently in the final design phase, with the main objectives of improving the control of magnetic confinement, plasma density and plasma wall interaction in both RFP and Tokamak configuration.
The achievement of these aims implies a major change and reconfiguration of the internal components of the machine assembly: the present toroidal support structure (Rmajor=2m, Rminor=0.5 m, made of stainless steel) will be adapted to provide the function of vacuum vessel and to encompass the conductive stabilizing shell, deeply modified in order to sustain the new first wall and a wide system of in-vessel diagnostics.
The machine modification is subject to stringent constraints in terms of geometrical interface with existing components (in particular external coils and diagnostic systems) and to the compliance with specified electrical insulation both along toroidal and poloidal direction, to allow suitable penetration of electromagnetic fields within the plasma chamber ant to minimize the risk of arcing among in-vessel components which could occur during over-voltages induced in standard or abnormal operating conditions.
The technical solutions conceived to fulfil geometrical, vacuum and electrical requirements have been developed in collaboration with some local industries, involved in the framework of an industrial innovation project supported by public regional funding. Critical aspects of the design are in particular: development of vacuum-tight electrical-insulated crossed joints of the new vessel, by using high-performance polymers; implementation of specific welding procedures to control the deformation of major machine components with tight tolerances; implementation of surface coatings of in-vacuum components; realization of in-vessel diagnostics and machine components by means of metal additive manufacturing processes.
The paper will present an overview of the design choices and the proposed implementations, assessed on the base of engineering analyses and results of experimental tests performed on mock-ups of the new components.

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P4.013 Effect of a magnetic cusp configuration on ion species ratio in a NBI ion source

In arc discharge plasma, an ion species ratio depends on plasma density and ion confinement time. To increase an atomic ion species ratio in an NBI (neutral beam injector) ion source, various ion sources have been designed to have higher ratio of the plasma volume to the effective ion loss area so that the ion confinement time could be longer. For that reason, most of NBI ion sources have a large arc chamber with strong cusp magnetic field. KSTAR NBI ion source also has a large arc chamber with a strong azimuthal line cusp configuration but its atomic ion (D+) beam fraction is small because its normal operation power levels of the arc discharges are comparatively lower than those of other NBI ion sources. Particularly, the D2+ (H2+) ion beam fraction is relatively large at about 40%. This means that there are so many primary electrons in the beam extraction region. Through several Ele-orbit code simulations, it was found that some magnetic cusp configurations are very effective to prevent primary electrons from getting near the plasma grid. In order to reduce the molecular ion fraction in KSTAR NBI ion source and consequently increase its atomic ion fraction, we carried out the experiments applying several new magnetic cusp configurations to the arc chamber. The results show that the atomic ion fraction could increases up to 80% with the new magnetic cusp configuration despite the low arc operation power at KSTAR NBI ion source.

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P4.015 System of 28 GHz ECHCD with beam focusing launcher on the QUEST spherical tokamak

Electron cyclotron heating and current drive (ECHCD) system with a 28 GHz gyrotron has been prepared for non-inductive electron cyclotron (EC) plasma ramp-up in the QUEST spherical tokamak (ST). Non-inductive plasma start-up using the EC waves is a key issue for advanced tokamak reactor concepts as well as for the ST concept. There are two important aspects of conducting the present ECHCD current ramp-up experiments. One is a beam focusing, and the other is incident polarization control. All elliptical polarization states can be controlled in combination with two corrugated directions of the polarizers with respect to incident planes of the waves. The two corrugated plates were designed and fabricated with careful attention to reduce Ohmic losses by means of high-precision milling, not wire-electrical discharge machining. The two-mirror launcher system has been developed to obtain a narrow beam size of w ~ 0.05 m. In the mirror surface design, the principle of the least propagating-phase was considered. The Kirchhoff integral and the Gaussian optics were used to evaluate the propagating-phases before and after the mirror reflection, respectively. The mirror area should cover a 1% intensity edge of 1.5w in the beam. The relatively large 2nd focusing-mirror with a diameter of 0.37 m was designed and fabricated for sharp beam focusing. The incident beam can be steered from perpendicular to tangential injections. The steering capability with excellent focusing property was confirmed at the low power test facilities. As a new record of non-inductive plasma ramp-up with EC-waves, a highest plasma current of 86 kA was achieved with a focused 230 kW 28GHz-beam. The record plasma current ramp-up efficiency on the incident power in the 2nd harmonic EC scenario was also achieved. The obtained electron density was one order of magnitude higher, compared to the previous experiments with no focusing-beam.

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P4.017 An overview of the in-vessel ICRF-diagnostics in the ASDEX-Upgrade tokamak

Radio Frequency (RF) waves in the Ion Cyclotron Range of Frequency (ICRF) are successfully used to heat fusion plasmas. For a better understanding of the ICRF wave propagation, absorption and other processes in the plasma good in-vessel diagnostics are essential.
In recent years, a number of high frequency B-field probes have been installed in the ASDEX Upgrade tokamak. These probes, arranged as an array, pick up the magnetic field components of the radiated wave, and can thus be used to characterize the radiation spectrum of the ICRF-antennas during plasma discharges. We have designed dual channel logarithmic RF-detectors with phase detection capability to track the fast variations of the probe signals during events like edge localized modes (ELMs).
The same probes are also used to measure the ion-cyclotron emission (ICE) of plasmas. A fast, open source data acquisition system that digitizes the rf-signal with 125 MSamples/s and 14 bit amplitude resolution records the ICE signals. To allow ICE measurements during ICRF operation notch filters are used to suppress the ICRF signal by 50 dB.
In order to characterize the drive mechanisms and consequences of RF sheaths on the structures around the ICRF-antennas, 12 antenna limiter tiles are equipped with shunts to allow the measurements of RF- and DC-currents during operation. RF-probes in the 3-strap ICRF antennas further provide information on the phasing between central and outer straps.

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P4.020 Design of the small scale prototype accelerator for KSTAR negative-ion-based NBI

The small scale prototype negative ion source has been designedfor KSTAR negative ion source. The target performance of the ion source is to extract 0.5 A of 200 keV D-. The aperture geometries of the accelerator gridsare based on the ITER HNBI reference design and optimized for the small

scale prototype ion source.The accelerator consists of plasma grid (PG), extraction grid (EG), and ground grid (GG). The nominal operation voltages of these grids are -200, -191, 0 kV for PG, EG, and GG, respectively. The extraction gap distance between PG and EG has been chosen as 6.0 mm, the same as ITER HNB extraction gap. 1-stage acceleration gap (between EG and GG) has been chosen as 100 mm. This is a compromise between high voltage holding of 200 kV and beam optics. Kerb plate (metallic plates around of the aperture) is also attached to the downstream surface of the extraction grid to compensate for the beamlet-beamlet interaction and to point the beamlets in the right direction. SmCo permanent magnets areemployed to suppress the co-extracted electron currents in the extraction grid. The design of the accelerator for the neutral beam injection on KSTAR is finished and fabrications are currently in progress. In this presentation, the considerations and choices which constitute the basis of the design are described.

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P4.022 Design of the Large Current Hydrogen Negative Ion Source by using Sheet Plasma

Production of negative ion plays an essential role in Neutral Beam Injection (NBI). Research on a cesium-free negative ion source using sheet plasma has being carried out. The sheet plasma is suitable to produce negative ions because the electron temperature in the central region of the plasma is as high as 10 -15 eV, whereas in the periphery of the plasma, a low temperature of a few eV is obtained. Therefore, it is considered that high density production of negative ion is possible in sheet plasma. The extraction of hydrogen negative ion beam from high density sheet plasma using TPD-Sheet IV had been succeeded. We have developed the large current hydrogen negative ion source to get more beam current from sheet plasma. The developed device has been increased in size to get negative ions from larger area of sheet plasma. The length of the extracting region is 132mm and width is 33mm. The device has 48 holes on the extracting electrodes. Since increasing of size of the device induces increasing of distance of magnetic coils, uniformity of magnetic field force is required. Therefore, the irons are put beside two coils which are both ends of extract device. The uniformity of magnetic field force is maintained about 10 % better than without irons. In addition, we have calculated the production of negative ions to set extract device at best region. From the calculated results of the production of negative ions using 0-dimentional rate balance equations, the extract device is set at about 14 mm above of center of the sheet plasma.

This study is supported by the LHD Joint Project of the National Institute for Fusion Science (NIFS16KOAR021).

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P4.023 The fabrication and assembly of the beam source for the SPIDER experiment

The ITER Heating Neutral Beam injector will be equipped with a beam source that will provide a negative beam of 40A (H2 or D2). The R&D activities undertaken in Europe to pursue this challenging goal comprise three experiments:
 ELISE the half size ion source experiment operating in IPP Garching
 SPIDER the full size ion source experiment at Neutral Beam Test Facility (NBTF) site in Padua
 MITICA the full size full energy ITER injector prototype being also established at the NBTF
The procurement of the beam source for SPIDER started in October 2012, when a contract was signed between Fusion for Energy and Thales Electron Devices (as group leader of a Consortium created with CECOM Srl, Galvano-T GmbH and Zanon SpA). The contract, still on-going at present stage, has been mostly completed in October 2017 with the delivery of the SPIDER beam source at the NBTF site in Padova Italy.

A review of the fabrication of the main parts and of the aspects of their assembly will be given with a focus on the difficulties encountered and the solutions adopted. The SPIDER beam source is in fact the first ITER full-size source and many of the components had never been manufactured before. Changes in the design induced by the results of the first experiment on the ELISE facility will be also described.
In the assembly phases of the project, some issues hampered the timely completion of the activities calling for delicate evaluations and the need of taking quick decisions. The paper will provide an overview of the risk management performed in support of the decision taken.
Finally, the description of the modifications adopted for the MITICA beam source that were identified during the manufacturing and assembly of the SPIDER beam source will be described in detail.

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P4.024 Electrical integration of two 1MW/2s dual-frequency gyrotrons into the EC-system of the TCV tokamak

The EC-system of the TCV tokamak was originally designed to accommodate nine gyrotrons, shared in three clusters of three gyrotrons; each cluster being supplied by one main high-voltage power supply [1]. As part of an important upgrade of TCV, this EC-system is currently renewed and extended [2]. Two MW-class dual-frequency gyrotrons (84 or 126GHz/2s/1MW) will be installed. The first one has already been delivered and will be fully commissioned on-site by the middle of 2018 [3].

This paper will describe the electrical integration of these two additional gyrotrons. The gyrotrons' cathodes will be fed by one of the main high-voltage power supplies, shared with the gyrotrons of an existing cluster. The commutation of the main power supply between the existing gyrotrons and the new ones will be performed remotely by the means of electromechanical switches. The anode power supply and the filament power supply will be integrated into a common shielded cabinet allowing the installation in a free access area. The electrical supply scheme complies with the installation principle selected for the EU gyrotrons at ITER, aiming to minimize electromagnetic interferences. The snubber used to damp the energy delivered to the gyrotron in case of an arc is implemented inside the gyrotron oil tank, as it is done in the EU gyrotron of the EC Test Facility in Lausanne [4]. The water cooling system will also be described, focusing on the electrical point of view, as well as the calorimetry measurements.

Based on long term experience of the Swiss Plasma Center with gyrotrons, giving special emphasis on electromagnetic compatibility, overall electrical design choices will ensure a reliable operation of these two gyrotrons.

[1] T. Goodmann, 19th SOFT, 1997, p. 565–568
[2] A. Fasoli, this conference
[3] S. Alberti, this conference
[4] D. Fasel, this conference

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P4.029 Enhanced operation of the EU EC Test Facility

A Test Facility (TF) has been designed and installed at SPC to allow for the commissioning of the EU gyrotrons developed in view of their integration to the ITER EC system. The first phase of operation of this TF was dedicated to the development of the EU 2MW coaxial cavity gyrotron [1,2]. The EU gyrotron development for ITER has been reoriented since then and is presently advancing a 170GHz/1MW/CW gyrotron based on a technical design similar to the 140GHz/0.9MW/CW operated at W7X.

This paper will describe first the technical modifications brought to the existing TF to comply with the requirements of the new EU 1MW gyrotron designed within the EGYC consortium and manufactured by THALES. Not only mechanical adaptations of the support tower have been implemented, but also a complete review of the cooling system and of the auxiliaries (SCM PS, filament PS etc.) has been implemented.
In parallel, this TF has been updated to include a second tower to host a 170GHz/1MW/CW gyrotron from GYCOM. This facility, named FALCON, is used to perform testing of the main RF components part of the EC transmission line and Upper Launcher system installed on ITER. Care has been taken to permit switching of the auxiliaries from one tower to the other within a short time (<1 week) while maintaining operator safety and equipment protection. A detailed description of the additional implementation will be presented, focusing on the strategy followed to share the main auxiliaries (like the HVPS, the cooling system, the control, etc.) in order to ensure a safe and optimized operation.
Moreover, most of the new equipment, such as the control hardware, has been designed to fulfil the ITER site applicable standard. Finally, the preliminary operation results with this reconfigured TF will be described.

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P4.031 New features of the ECH control system in DIII-D and impact of power modulation on collector loading

The electron cyclotron heating system in DIII-D comprises four 110-GHz gyrotrons and one 117.5-GHz gyrotron able to inject more than 3 MW into the plasma for an administrative limit of 5 s during the 2018 experimental campaign. In a major controls upgrade, the gyrotron high voltage reference waveform is no longer generated by an obsolete pre-programmed waveform generator but by a newly developed adaptive FPGA-based generator. This new generator outputs 16-bit signals every microsecond and responds in real time, being able to detect, block faults, and restart full RF power within 10 ms during plasma or conditioning shots. Some gyrotron faults, such as excessive body current, can be detected and avoided within 10 μs by the new system.
In order to control plasma instabilities, gyrotrons are required to provide modulated power with frequencies up to a few kilohertz. Power modulation can be achieved by reducing the main cathode voltage while keeping the body deceleration voltage constant. The impact of modulating and operating with reduced interaction efficiency and lower electron beam power on collector loading has been quantified using a newly developed and improved collector mapping system.
A description of the mapping system along with collector loading results will be presented. The performance of the integrated real-time FPGA-based waveform generator during gyrotron faults and over-dense plasmas as well as upgrades to the ECH data acquisition system will also be presented.

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P4.034 Modeling of ITER initiation scenarios using IMAS framework

A careful control of poloidal field (PF) coil currents is indispensable to assure a successful plasma initiation in ITER. This requires the development of an accurate modeling tool which can evaluate PF coil current and voltage waveforms leading to a satisfactory breakdown condition. In particular, it is of most importance to provide the necessary loop voltage with a sufficiently large field null in the presence of a large amount of eddy current anticipated in ITER plasma initiation. In this study, we develop an ITER plasma initiation scenario by using a two-dimensional ITER conductor model. The realistic power supply constraints such as the maximum voltage and current are taken into account. The constraints for flux and fields are imposed by using a constrained least square method. We

evaluate various scenarios and provide an optimal initiation scenario, satisfying all the constraints imposed to obtain a successful breakdown. In addition to this, we identify the most vulnerable point in terms of inductive heating for each PF coil. This information will be of importance for a thermos-hydraulic analysis of ITER PF coils. The simulation code is developed using the ITER Integrated Modeling Analysis Suite (IMAS) data structure for an easy adaptation to the integrated ITER simulator.

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P4.036 Real Time Assessment of the Magnetic Diagnostic System in RFX-mod

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The magnetic diagnostic system in RFX-mod [1] includes local field probes, flux loops and saddle loops. It plays an important role in real time plasma control and in several off line studies of the plasma configurations and behaviours. The failure or malfunction of one or more components can cause negative consequences in several applications, including the plasma equilibrium and stability control.
Unfortunately, the accessibility of the diagnostic components is rather difficult; therefore, their assessment in RFX-mod is carried out by suitably operating the devices in dry discharges or in well controlled plasma conditions and suitable calibration coefficients are introduced.
Recently, a new procedure has been proposed [2] able to identify faults in a magnetic diagnostic system, to provide each component with a reliable confidence coefficient and to suggest reasonable corrections of the actual measuring. The procedure is able to operate also in presence of plasma. The methodology is based on a "mirror" operator performing two main actions; a first, inverse action models the unknown sources including the plasma equivalent currents and a latter, direct action reconstructs the magnetic field in the region of interest. The capability of the representation in the inverse step can be suitably calibrated to bring out, in the second direct step, the discrepancies to be interpreted as faults or malfunctioning.
The paper describes the use of the methodology to the RFX-mod and, in addition, discusses the performance achieved also in shots with fully 3D plasmas.

[1] P. Bettini, A.G. Chiariello, A. Formisano, F. Ledda, G. Marchiori, R. Martone, F. Pizzo, D. Terranova. "3D magnetic surfaces reconstruction in RFX-mod". Fusion Engineering and Design, 123, 546-550. (2017).
[2] A. Chiariello, A. Formisano, F. Ledda, R. Martone, F. Pizzo. "A Fast Reconstruction Approach for the Assessment of Magnetic Diagnostic Systems in Nuclear Fusion Devices". IEEE Tr MAG, DOI 10.1109/TMAG.2017.2771753

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P4.041 Plasma shape control assessment for JT-60SA using the CREATE tools

The Satellite Tokamak Programme (STP) is the main project within the Broader Approach agreement. The STP includes the construction of the JT-60SA superconductive tokamak and its exploitation as an ITER "satellite" facility. In view of JT-60SA operations, Japanese and European scientists are developing different tools to support preliminary studies. In this context, a set of tools for the design and the validation of plasma magnetic control have been developed. Indeed, plasma magnetic control is needed since early tokamak operations to drive the currents in the external active coils, in

order to achieve plasma breakdown and to track the scenario current waveforms. The CREATE electromagnetic modelling tools were used to design and validate a set of magnetic control algorithms for JT-60SA in an already proposed architecture for the magnetic control system.

A gap-based algorithm for plasma shape control is introduced in this paper. The proposed approach is based on the eXtreme Shape Controller, which is currently used at the JET tokamak. JT-60SA represents a relevant benchmark to further validate this control approach given the high beta regimes that are envisaged during its operation. Indeed, such regimes represent a challenge from the plasma magnetic control perspective.

Different test scenarios will be considered to assess the performance of the proposed shape controller, with the aim of defining an optimal set of gaps to be controlled. The capability of tracking different plasma shapes, as well as the one of rejecting the envisaged disturbances is considered. The result of this analysis suggests that a set of about 20 gaps equally spaced along the plasma boundary permits to control the shape with a root mean square error (computed on about 80 gaps) less than 1 cm, at steady state, when the specific disturbances for the flattop are considered.

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P4.039 Robust control of the current profile and plasma energy in EAST

Integrated control of the toroidal current density profile, or alternatively the q-profile, and plasma stored energy is essential to achieve advanced plasma scenarios characterized by high plasma confinement, magnetohydrodynamics stability, and noninductively driven plasma current. The q-profile evolution is closely related to the evolution of the poloidal magnetic flux profile, whose dynamics is modeled by a nonlinear partial differential equation (PDE) referred to as the magnetic-flux diffusion equation (MDE). The MDE prediction depends heavily on the chosen models for the electron temperature, plasma resistivity, and non-inductive current drives. To aid controller synthesis, control-oriented models for these plasma quantities are necessary to make the problem tractable. However, a relatively large deviation between the predictions by these control-oriented models and experimental data is not uncommon. For this reason, the electron temperature, plasma resistivity, and non-inductive current drives are modeled in this work as the product of an "uncertain" reference profile and a nonlinear function of the different auxiliary heating and current-drive (H&CD) source powers and the total plasma current. The uncertainties are quantified in such a way that the family of models arising from the modeling process is able to capture the q-profile and plasma stored energy dynamics from a typical EAST shot. A control-oriented nonlinear PDE model is developed by combining the MDE with the "uncertain" models for the electron temperature, plasma resistivity, and non-inductive current drives. This model is then rewritten into a control framework to design a controller that is robust against the modeled uncertainties. The resulting controller utilizes EAST's H&CD powers and total plasma current to regulate the q-profile and plasma stored energy even when mismatches between modeled and actual dynamics are present. The effectiveness of the controller is demonstrated through nonlinear simulations.

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P4.044 Operational experience And evolution of the Real Time Protection Sequencer on JET

The Real Time Protection Sequencer allows the Session Leader to program magnetic and kinetic actuators in response to alarms. We will deal with the State Diagram, actuator conflicts and Jump to Termination, which drove the changes to the software and architecture in [1].

The State Diagram determines what happens when one response type follows another, e.g a main chamber hot spot response is followed by a request for a divertor hot spot response and ensures that this is re-directed to a combined response type. The State Diagram is defined in a transition table and is changeable to correct flaws or meet new experimental requirements, as will be shown.
Actuator conflicts arise from the integration with older systems, but were not resolved. This is now done by defining rules which identify such conflicts at the pulse design stage and guide the SL towards a valid set up. This move from the real time code to the pre-pulse preparation and validation will be discussed.
Jump To Termination (JTT) allows us to jump forward to a preset time in the pulse in response to an alarm. This is a difficult to implement on JET due to the Central Timing System and the distributed nature of control. All subsystems and operators have to understand JTT in the same way. It has proved extremely powerful, but there are pitfalls. It can only be the first response, the transitions are the same as if reached naturally, and subsequent stops take their definitions from the correct time window.
[1] J Edwards ibid.
This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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P4.046 Runaway electrons diagnostics using segmented semiconductor detectors

The runaway electron generation during plasma disruptions presents a danger to the vacuum vessel and associated instrumentation. The presented work concerns application of semiconductor detectors for study and characterization of runaway electrons events. The recent advances in the field of semiconductor detectors allow for the development of new diagnostic methods, utilizing their particle detection capabilities,
including unprecedented dynamic range as well as spatial and temporal resolutions. They can be applied as diagnostic instruments, enabling better resolution of both direct measurement of charged particles and indirect one of X-rays or neutrons. In tokamaks, both processes could be exploited for detecting runaway electrons - either directly inside the vacuum chamber, or indirectly via radiation produced by interaction with plasma facing material. Results of the new technique are to be compared with already existing diagnostic methods such as scintillation devices detecting X-rays and photoneutrons.

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P4.049 Effect of temperature on magnetic measurements in KSTAR tokamak

KSTAR plasma experiments have been carried out without baking the inner wall (called as cold wall condition) until 2016. The drift (ΔVsen/Vsen) in magnetic sensor signal was able to satisfy with the requirement for the plasma control by adjusting the offset level of the input in the integrator for several the vacuum shots. Here, the required value of (ΔVsen/Vsen) should be below 2 % during 60 s.
However, plasma experiments are performed after the temperature on the inner wall reaches up to 150 ° (called as hot wall condition) in the experimental campaign of 2017 in order to achieve better plasma performance due to lower recycling from the wall during a plasma discharge.
As the temperature of the magnetic sensor increases, the drift of the integrator also increases and

exceeds the desired limit above. Hence, adjusting the offset was performed again with hot wall condition to achieve the previously desired limit for plasma control. Consequently, in the 2017 plasma experiment the drift variation did not differ significantly between hot and cold wall conditions, even up to 100 s.
In this presentation, we explain the measures for the drift of the integrator by hot wall and cold wall.

*This work was supported by the Korea Ministry of Science, ICT and Future Planning under the KSTAR project contract.

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P4.054 On multistage development of ITER High Field Side Reflectometry diagnostic module design based on thermal stress numerical assessment

The High Field Side Reflectometry (HFSR) is one of the ITER diagnostics, which provides information about plasma state by measuring the electron density profile. HFSR diagnostics undergoes a strong action of different physical nature loads.

Five various designs have been developed and studied since 2014 as a result of interaction of Peter the Great St. Petersburg Polytechnic University, NRC "Kurchatov Institute" and Fusion Centre. Multiphysics engineering finite-element analysis has been performed for each of the HFSR design.

This study deals with thermal loads which were discovered to be the most significant from structural integrity point of view. HFSR is subjected to non-uniform nuclear heating during plasma operation, high-intensity antenna radiative heating from plasma and substantial cyclic thermal loads during ITER life that makes satisfying the strength requirements challenging. Several thermal analyses were conducted to determine conditions ensuring the least and the most dangerous state.

Key features of the proposed mathematical model and results of numerical simulations which demonstrate a stress state of the diagnostics are given. The process of multistage diagnostic design development based on the finite element analysis results is presented. In addition, the study of bolt tightening force and friction coefficient values influence on a stress-strain state of the final diagnostic design is fulfilled. The necessity of accounting these factors in numerical simulations is demonstrated.

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P4.057 Plasma cleaning of ITER diagnostic mirrors with coaxial λ/4 water-cooling line

First mirrors (FMs) of optical diagnostics in ITER will operate in a harsh environmental conditions. Deposition of plasma-facing materials on the mirror surface will cause a degradation of the nominal diagnostic performance. During ITER lifetime the mirror optical characteristics are intended to be periodically recovered by an appropriate cleaning technique. Cleaning system based on the RF capacitively-coupled discharge (RFCCD) with a mirror used as a RF-electrode is currently considered as the most promising technique. FMs in ITER are subject to high heat loads due to neutron flux and the intense radiation. To prevent overheating, most FMs are water-cooled. Tests of the mirror cleaning system concept with water-cooling line designed as RF notch filter were performed. The notch filter performance was verified for D2 and noble discharges at the pressure of few Pa. The λ/4 shorted coaxial mineral cable attached to the RF-electrode was used as an electrical analogue of the cooling line. The IEDF and ion flux measured at the powered electrode in the schemes with and without stop band filter were compared. The concept of cooling line integrated into notch filter

was analyzed by assessment of Be/BeO/W sputtering rates for different operating parameters e.g. pressure, gas type, RF power etc. The sputtering efficiency and homogeneity were studied at the frequency of 81 MHz for two typical cleaning scenarios: with and without magnetic field (up to 0.7 T) in the experiments on Mo mirror cleaning. The scaled laboratory mock-up of the edge CXRS cooled FM with water line designed as λ/4 stop band filter was used for the test cleaning from proxy metal coatings. Acknowledgement: This report supported in part by ITER Organization (contract –IO/17/CT/4300001626) was prepared as an account of work for the IO. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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P4.061 Steady-state magnetic diagnostic for ITER and beyond

Magnetic measurements at long pulse magnetic confinement fusion devices require implementation of the true steady state magnetic field sensors in order to achieve required precision of plasma position measurement. Inductive sensors can suffer from a range of temperature gradient and radiation induced offsets which together with the intrinsic offsets of analogue integrators can lead to unwanted artificial drifts of their output signals.
This contribution will present overview of the several years long R&D programme dedicated to development and qualification of the outer vessel steady state magnetic sensors based on bismuth Hall sensors. The main results regarding sensor design and optimization, technology of manufacturing, results of number of qualification tests including neutron irradiation tests, measurements of sensitivity versus temperature and magnetic field will be briefly reviewed. Elaborated design of the sensors housing which include temperature measurement thermocouple in-situ recalibration feature will be presented. Concept of Hall sensor controller which integrates synchronous approach to detection, current spinning technique to eliminate planar Hall voltage and offset and use of high reliability electronic components to assure long term stability and high accuracy will be presented. Finally, the current status of the manufacturing of the diagnostics, including lessons learned, will be given. Final part of the contribution will provide outlook of potential, challenges, requirements and new concepts of implementation of steady state magnetic sensors based on Hall effect in even more hostile environment of DEMO and future power reactors. The results of online investigation of Hall sensors based on gold nanofilms up to DEMO-relevant neutron fluences will be presented. New sensing materials like mixtures of bismuth and antimony/copper will be introduced.

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P4.080 A 15kA solid circuit breaker for the switch network unit of the EAST Tokamak

This paper describes the development and testing of a 15 kA Solid Circuit Breaker (SCB) applied to the Switch Network Unit (SNU) of the Experimental Advanced Superconducting Tokamak (EAST). The circuit scheme composed of parallel connected integrated gate-commutated thyristors (IGCT) and the diode bridge to realize bidirectional breaker ability is presented firstly. In this paper, the stray inductances of the bus bars and the heat sinks for the current sharing of each IGCT are considered in detail. Through Ansys Q3D parameters extraction, the influence of structure stray parameters on the current sharing of power device in parallel is analyzed. Finally, the operation of the SCB has been successfully tested up to the current of 15 kA and the maximum interruption voltage of about 2.4 kV, these experiment results confirmed the current balance of the parallel-connected devices and the reliability of the SCB during current conduction and current commutation processes.

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P4.065 The evaluation of server hardware for the fast online processing of X-ray impurities diagnostics in the WEST

The WEST thermal fusion reactor is currently an upgraded evaluation environment for the diagnostics to-be-deployed in the constructed ITER tokamak. In particular, fast metallic impurities diagnostics is tested, that in future can increase reaction efficiency and provide safe mode of operation with the divertor. A diagnostic system has been provided in the framework of the collaboration and the optimization of the processing within this instrument is ongoing. This is to reduce the latencies as far as possible in order to provide a feedback loop to the tokamak control systems. The heterogeneous diagnostic system implements an FPGA front-end and server back-end architecture, where computational workloads are decomposed into these devices.

Although the operational device has already been constructed with an FPGA front- end and server back-end, there is still an ongoing study how to augment and further optimize the back-end for new online diagnostic workloads.

This work discusses the tests of selected hardware accelerators on the back-end server PC side, in particular, the PCIe cards and CPUs. Their feasibility for the application in the WEST is evaluated and assessed. The bottlenecks of implementations are discussed and the issues of provision of runtime processing are presented. The relevant hardware limitations are pinpointed. The performance results of the runs of the optimized algorithms with the accelerators are shown. This is to both maximize the throughput of data processing from a multichannel, multipixel GEM detector for the calculation of spatial distribution of impurities in the running experiment in the reactor.

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P4.069 Structural Thermal Optical Performance analysis for full system optical performance assessment in complex environments

Optical design is a discipline where craftsmen assemble optical systems making use of design history and analogy to other systems, often followed by brute force optimization. In the subsequent design phase, a mechanical design is created to hold the optical elements in place and to provide adjustment mechanisms. Using this design knowledge an initial performance prediction can be made of the assembled opto-mechanical system. The impact of any environmental influence like gravity or thermal distribution often is estimated using statistical analysis of assumed perturbations or by partial analysis of deformed optical analysis. For optical systems which require ultimate performance, or which are operated under harsh environments this is a very crude and inaccurate approach.
TNO has developed a method for full system optical performance analysis where the complex influences introduced by its environment are incorporated in an accurate, and repeatable manner with built-in verifications. This Structural Thermal Optical Performance (STOP) Analysis provides a direct and reproducible link between the structural thermal analysis for combined load cases comprising nuclear heating, cooling water pressure and temperature, gravity and mounting stresses (in Ansys) and the associated changes of the optical performance, including imaging quality and alignment (in Zemax OpticStudio). This enables assessing the impact of a variety of load cases on optical performance.
This new analysis process is demonstrated on the Upper port Wide Angle Viewing System in the ITER tokamak as well as on the telescopes and UV1 spectrometer of the Sentinel 5 instrument in the EUMETSAT environment monitoring satellite. Here up to 80 different operational load cases are analyzed to accurately assess the optical performance during the various phases of the system operation.

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07NA27344.

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P4.070 Performance evaluation of a hypothetical core density fluctuation measurement on the ITER diagnostic beam.

Understanding core MHD activities is of particular interest on ITER. The fast measurement of core density fluctuations might be a viable avenue for the exploration of plasma core activities in some of the ITER operational scenarios. Fluctuation beam emission spectroscopy (BES) would utilize the diagnostic neutral beam shot into the plasma, where beam atoms get to excited states due to collisional processes, and cause spontaneous light emission.

The feasibility of a pedestal density fluctuation measurement by BES on the ITER diagnostic beam has been recently demonstrated for the observation of edge turbulence and MHD activities [1], and previous work indicated promising results for a core density fluctuation measurement concept [2]. Aim of present contribution is to evaluate the expected performance of a hypothetical system that would operate in conjunction with the core charge exchange recombination spectroscopy system located on upper port 3 [3].

Forward modelling was used to estimate the signal to background and signal to noise ratios, as well as the spatial resolution of the hypothetical system by use the RENATE 3D BES modelling code, which handles realistic magnetic geometries and accounts for the spatial effects of the diagnostic. Simulation of Spectra was used to calculate spectrum of the beam emission, and estimate the corresponding theoretical filter throughput. Fluctuation response analysis was performed on the system and evaluated in view of possible applications of the diagnostic with regard to observable density fluctuation amplitudes and mode numbers.

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

[1] O. Asztalos et.al. EPS submitted to (2018)
[2] G.I. Pokol et.al Fusion Eng. Des. 88 1386 (2013)
[3] A. Krimmer et.al. SOFT submitted to (2018)

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P4.072 Electromagnetic loads on ITER low field Side reflectometer in-vessel components

The current design for the ITER Low Field Side Reflectometer (LFSR) diagnostic system contains six double-walled circular waveguides that function as both launch and receive antennas. The waveguides run from the diagnostic first wall (DFW) to the source/detector with the total length of approximately 40 meters. The front end of the LFSR, interfacing with the plasma facing DFW and its supporting diagnostic shielding Modules (DSM), is integrated into equatorial port plug 11 (EPP #11) by the Russian Federal Domestic Agency (RFDA). The LFSR components are required to function in a high temperature, high electromagnetic (EM) field and high nuclear radiation environment in the port zone.
The EM loads are body forces on the metallic components resulting from interaction of eddy currents, induced by the fast plasma current disruption, in the conductive LFSR and port components, with the background magnetic fields from TF, PF and central solenoid (CS) coils. The induced eddy currents in the components are calculated using 68 toroidal filaments on the toroidal perimeter of the plasma, carrying time-varying currents to mimic the ITER current transients. Based on 2D DINA results, and the results of prior full 3D transient simulations of the several worst DINA cases chosen

from a 2D DINA scan, a major disruption with downward vertical drift and 16ms exponential current decay time, MDDWEXP16, was considered the most severe scenario for the EPP whole port as well as for the DSMs and the LFSR components therein. The body force density on the conducting structures were mapped onto the finite-element model for structural analyses (coupled with thermal and thermal hydraulic analyses) of the in-vessel components. The effect of the EM laods on the front end components of the LFSR was shown to be small compared with thermal loads. The results of this modeling are presented here.

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P4.086 HTS CroCo – Demonstration of long length fabrication

High-temperature superconductor (HTS) CrossConductors (HTS CroCo) are twisted stacked strands built from HTS tapes, optimized for high engineering current density and easy long length production [1,2].
The production for a 35 kA DC cable demonstrator required increased amounts of HTS CroCos for which the production of HTS CroCos was extended to 8 m length. This milestone shows clearly that an extension to even longer lengths is feasible, assuming that tapes of corresponding length (e.g. > 100 m) are easily available.
This achievement demonstrates, that the HTS CroCo is a basic strand, that allows an easy production of high current HTS strands, that can be used for high current power transfer or for large magnets which are based on high current conductors.
The contribution will highlight the long length production of HTS CroCo strands, pointing to future application in power transfer and high field magnet application.

[1] M. J. Wolf et al., IEEE TAS 26 (2) (2016), 6400106
[2] W. H. Fietz, et al., IEEE TAS 26 (4) (2016) 4800705

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P4.094 Mechanical analysis of the European DEMO central solenoid pre-load structure and coils

The pre-conceptual design of the European DEMO fusion tokamak is currently being developed under the coordination of the EUROfusion Consortium. This paper reports the mechanical analysis of the central solenoid (CS), which comes right after the phase of definition of the winding pack proposed by the CEA. An analytical model is firstly developed in order to approximate the required axial preload level, considering the electromagnetic forces and the cool-down thermal shrinkage of the coils. This first step allows to define the tie-plates pre-compression structure. Then a global FEA of the CS, considering a sector-symmetry model and smeared winding pack properties is built-up in order to check the pre-load structure, and to identify the most critical loading scenario for the conductor. Finally, an axisymmetric local FEA of the CS winding pack is performed in order to study in detail the stress level in the conductor jacket and in the insulations layers. This procedure allowed to assess the acceptability of stress for the CEA winding pack proposal, and will be will be operated during the future magnetic system design iterations.

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P4.073 Doppler coherence imaging of impurity flows in the divertor of tokamaks and stellarators

In magnetically confining plasma experiments, measurement of ion dynamics is of great importance to study the impurity transport in the scrape-off-layer (SOL) and divertor. Impurities from the divertor areas may degrade the core energy confinement, if transported too far upstream in the SOL. The Doppler coherence imaging spectroscopy (CIS) is a relatively new technique [1] for the observation of ion and neutral flows. It is a passive optical diagnostic which produces 2D images of line-integrated measurements of the ion flow or ion temperature. Doppler CIS measurements conducted in the poloidal field divertor of the medium-sized tokamak experiment ASDEX-Upgrade and the optimized stellarator Wendelstein 7-X reveal near-sonic (Mach ~ 1) impurity ion flows in the divertor. The SOL and divertor impurity ion flows in tokamaks and stellarator SOL are subject to many forces [2]: they can either be transported downstream by e.g. friction with the main plasma flow, but also upstream due to thermal gradient forces, for example. In AUG, the impurity flows measured with the Doppler CIS were observed to be parallel to the magnetic field lines, pointing away from the divertor target plates. In W7-X, perpendicular transport is expected to play a larger role [3] and first measurements have been conducted.
Both Doppler CIS measurements in AUG and W7-X are aimed at a fundamental understanding of the impurity SOL transport in different divertor regimes and to determine operation scenarios that prevent impurity transport into the core, which is also relevant for e.g. ITER. A new method was used to gain absolutely calibrated flow measurements in AUG and W7-X and will be presented, along with first measurement results and interpretation.

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P4.076 Extension of the high current power supplies of ASDEX Upgrade – The design of a new thyristor converter

Due to the successful program of ASDEX Upgrade (AUG) the high current converters come to their limits. At the same time the risk of failure increases because of ageing. At the moment all converters are in use and thus the area of operations is fixed. Low power, high density discharges of AUG are limited by the OH transformer flux. Therefore a new thyristor converter group, called “Group 7”, is planned.
In combination with an existing converter (called “Group 6”), the OH supply will be increased from +/-40 kA / 5 s to +/-45 kA / 10 s for each polarity.
The new converter includes the so called “phase-to-neutral voltage control” with 2 thyristors more for the star point of the converter transformer, which allows a substantial reduction of the reactive power demand under partial-load operation.
While the design of the power section (transformers, thyristor-stacks, dc-chokes, …) will be in most parts a copy of Group 6, the control system will be a completely new design.
Group 7 gets a fully digital control system, which will increase the reliability and simplify the design using more standard components. While fast control issues like the control loops and the firing pulse generation are processed on a FPGA, less time-critical tasks will be done one a PLC (programmable logic controller).
To integrate this digital control system, also the interface to the AUG control system will be transformed from an analog plug, to a mostly digital interface via Ethernet.
The design phase is currently ongoing and in parallel preliminary works to integrate the new converter in our power supply network, like the extension of the 10kV-switchgear, the thyristor-crowbar-system and the crossbar-distributor, started.

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P4.077 Venturi flowmeters for the control of the ITER magnets

Within the framework of the ITER project, CEA/SBT is in charge of the design, manufacturing and delivery of 277 Venturi tube flowmeters for the control of the superconducting magnets. Six types of flowmeters were developed for either the control of the supercritical helium flow in the magnets at cryogenic temperature (4.5 K) or the operation of the current leads at room temperature. The last flowmeters were delivered to the ITER organisation this year, after four years of work.
The values of the Reynolds number in these measuring devices are unusual and outside the range of the standard NF EN ISO 5167-4 regarding the design of Venturi flowmeters. The flowmeters installed at room temperature are used under laminar and turbulent flow regimes, while those installed at cryogenic temperature are used under turbulent flow regime with high Reynolds numbers.
In this context, an experimental determination of the flowmeter coefficient was required. As a result of the large number of flowmeters, a manufacturing strategy was developed in order to obtain a reproducible behaviour from one type of flowmeter to another. The test benches used for the characterisation of the flowmeters as well as the experimental results that validated this strategy are presented.

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P4.078 Implementation Details of an Upgrade to the EAST Poloidal Field Power Supply DAQ System

The Experimental Advanced Superconducting Tokamak (EAST) poloidal Field Power Supply has recently implemented the upgrade of the control system. Inspired by the ITER Control Data Access and Communication (CODAC) system, experimental physics and industrial control system (EPICS) has been chosen for the control system. The Data Acquisition (DAQ) system is an important subsystem of the overall control system. This DAQ upgrade is bases on the EPICS design to meet the requirements of long pulse discharge within the tokamak and be consistent with the existing control system. The I/O interrupt mechanism on the device support module in EPICS is used to read changed data, which reduces the network communication load. The asynDriver is used to develop the device specific interfacing code for the DAQ card driver. The benefits of this upgrade include a significant reduction in the number of source files and lines of code as well as better trace and debugging control. The DAQ system has had good performance during the experiments and convenient human-machine interface to satisfy the requirements of all the experiments and experimenters. This paper describes the details of implementations

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P4.079 Research on Synchronization Technology of High-power Converters

In the Poloidal Field (PF) Power-supply System of International Thermonuclear Experimental Reactor (ITER), the phase-control thyristor converter is used to supply power to the PF superconducting coil. A precise zero scale is provided for the thyristor trigger by synchronization technology. The method of extracting the synchronization signal on the rectifier transformer primary side is widely used. However, the operation performance of the ITER PF converter is seriously affected by the dynamic phase difference because of the leakage inductance of the rectifier transformer. An approach to extract the synchronization signal on the secondary side of the rectifier transformer is proposed.

Then, an improved three-phase phase-locked loop method named frequency adaptive moving average filter PLL (FAMAF-PLL) is proposed to eliminate the effect of the unbalanced voltage and characteristic harmonics. This method can accurately realize real-time synchronization and effectively improve the PF converter performance. Finally, the reliability and accuracy of the proposed synchronization technology is verified by simulation and experimental results.

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P4.082 Design and optimization of vacuum pressure impregnation mould for large scale superconducting coils

The Poloidal Field (PF) coils are one of the main sub-systems of the ITER magnets. Fusion for energy (F4E) is responsible for supply five of them(PF2-PF6) as in-kind contributions to ITER project. ITER PF6 coil is being manufactured by the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP) as per the Poloidal Field coils cooperation agreement signed between ASIPP and F4E.
The technology of design and construction of superconducting coils has evolved dramatically over the last decades. The very large and complexity coils are demanded by fusion technology, where the Vacuum Pressure Impregnation (VPI) technique has become the most common process for the consolidating electrical insulation of large superconducting coils. To achieve success with the VPI process, superconducting coil is contained within a vacuum chamber of VPI mould to prevent resin leakage under operating conditions. This paper is focusing on the design and optimization of VPI mould of the mock-up of the pre-winding pancakes (PWPs), one-tenth of the ITER PF6 winding pancake (WP). The mock-up weighs 30 tons and is 1.2 m high, its VPI mould need meet three main requirements. 1) adequate structure strength, 2) pumping for coil de-gasing at 110 ℃ and resin curing at 130 ℃, 3) using over-pressure to force resin into the coil structure. Using structure mechanics and thermodynamic analysis with finite element (FE) to optimize the VPI mould model. In addition, getting experimental figures of the VPI mould from actual VPI process of the mock-up, to compare with the optimized VPI mould model, to optimize the VPI mould further. The results show that VPI mould of the mock-up can meet requirements of VPI process.

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P4.084 Design a Suitable Test Scheme for Triggering Bypass Protection Test of ITER PF Converter Unit

The external bypass, as an important components of the international thermonuclear experimental reactor (ITER) poloidal field converter unit (PFCU), will provide a freewheeling loop for the load current to protect the magnets and PF converter modules from being damaged by over-current and over-voltage under fault conditions. The triggering bypass protection test is used to verify that the designed bypass is triggered normally and endure the rated load current. In this paper, a suitable test scheme is designed for triggering bypass protection test of ITER PFCU based on the PF converter integrated test platform at ASIPP. This test scheme includes the triggering bypass method, calculating the trigger angle of the converter in inverter and a method of reducing the bypass current to zero in the absence of ITER mechanical switch. The feasibility of the test scheme is successfully verified by the simulation results and test results, both of which prove that the external bypass of ITER PFCU is triggered normally and withstands the load current for 100ms. Due to the integrated inductance of the actual DC output circuit, the actual transfer time of bypass current in the test is more than the simulation results.

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P4.085 Investigations on stable operational regions for SPIDER RF oscillators

The SPIDER experiment, currently starting to operate at the Neutral Beam Test Facility (NBTF) in Padova, is the full-size prototype of the radiofrequency (RF) ion source for ITER Neutral Beam Injector (NBI). It features a 1MHz RF system including 4 generators, each one rated for delivering up to 200kW of RF active power. A single generator is composed of 2 tetrodes connected with the push-pull configuration. Its load is made of a 30m long coaxial transmission line plus the plasma driver with its matching box. The selection of free-excited oscillators for the RF generators of ITER NBI ion sources was done on the basis of the experience at IPP laboratory suggesting to prefer that technology since it allows operation against variation of the load over a wide range. However, more recent experience at IPP reports on the observation of so-called "frequency-flip" phenomenon, i.e. a sudden frequency variation of the oscillator. Moreover, the site commissioning tests of the SPIDER-RF generators showed that, under certain mismatched load conditions, an amplitude modulation of the fundamental harmonic rose up leading to unstable operation. So, studies have been carried out in order to understand the causes of such instabilities. Three models have been developed for the SPIDER-RF system: a full transient model, a phasor-based model and a state-space model with eigenvalue analysis. With the aid of such complete set of models, it was possible to reproduce, understand and fix the causes of the modulation. Moreover, the models predict that also SPIDER could be affected by the "frequency-flip". Nevertheless, the eigenvalues analysis explains the causes of the flips and the phasor model defines safe operational regions for the generators. The paper will report on the studies and models development to understand the behaviour of SPIDER-RF oscillators and to investigate on the achievement of stable operating conditions.

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P4.088 AC loss and stability analysis of EAST superconducting magnet in ELM triggering

The ELM triggering with H-mode will cause energy loss of plasma in Tokamak device. The phenomenon results in the timely response of control system and motivates the rapid variation of current in PF coils which affects the AC loss and stability of superconducting magnet. The cryogenic parameters of the superconducting magnet will be analyzed according to the EAST experimental data for recent years in this paper. The AC loss of superconducting magnet in ELM triggering will be acquired to determine the stability margin in operation temperature.

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P4.090 Development of Supervisory Control System for Magnet Power Supplies in JT-60SA

Magnet power supply (PS) system of JT-60SA is being implemented within the Broader Approach Agreement aiming to achieve first plasma in 2020. The system consists of several components such as DC power supply, switching network unit, quench protection circuit and motor-generator. All the components have their own internal controller (LCC) for stand-alone operation so the individual commissioning are in progress and almost completed.
The next step scheduled is a combination test with all components in which supervisory control system manage LCCs remotely. The supervisory system covers the following roles: 1) machine status monitoring; 2) discharge sequence management; 3) real-time reference waveforms management;

4) data acquisition; 5) machine protection; 6) human safety. The supervisory system consists of several subsystems; PS supervising computer (PS-SC), PS programmable logic controller (PS-PLC), PS internal protection system(PS-IPS) and PS safety interlock system (PS-SIS).
PS-SC covers the primary roles in 1) – 4) which is built on CompactPCIs with real-time OS INtime. PS-SC and LCCs are linked through a Reflective Memory (RM) network to share common data in real-time (4 kHz). PS-SC is also linked to the upstream JT-60SA supervisory control system (SCSDAS) via another RM network as an interface for the power supply internal control systems. During discharge sequence, PS-SC also manages motor-generator speed, AC circuit breakers OPEN/CLOSE status with a controller PS-PLC and magnets current according to request from SCSDAS.
PS-IPS covers the function of 5) which is implemented on FPGA. PS-IPS is connected to LCCs via fiber optical and hard-wired signals. In case of emergency events such as magnet quench and earth quake, PS-IPS commands to shut down the magnets currents immediately.
PS-SIS plays the role in 6) such as status and permission of fence OPEN/CLOSE and grounding switch for maintenance.
In this presentation, overview and development of the supervisory system are described.

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P4.096 Ultrasonic C-scanning technology application for low-temperature superconducting joint box material of ITER current lead

The low-temperature superconducting (LTS) joint box is an important part of ITER HTS current leads. Enabling to provide the required functionality, the LTS joint boxes are made out of Copper-316L bi-metallic explosion plates. The bimetal interface of the joint has the direct effect on the mechanical properties of the joint and testing performance at low temperature. For this reason, the interface of the material must be of the highest quality without internal flaws.
This paper describes work on the development of water immersion ultrasonic C-scanning technology, and its application on bi-metallic explosion plate. It also give a comparing results on detecting the line segments and arc segments from copper side by using single crystal focusing probe and linear phased array probe. Assessment criteria were defined by analyzing signals which is produced by interface and the defects. Experimental results indicate that the method can meet the requirement as mentioned that the absence of defects of area greater than $2mm^2$ in the final box should be confirmed.

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P4.091 MESS: a new Magnetic Energy Storage Scheme for improving the power handling in fusion experiments a conceptual study

The plasma breakdown and ramp up in fusion experiments require high peaks of power, increasing with the size of the machines and plasma current value. In ITER, the power peak reaches 600 MW and in DEMO the present estimation gives values higher than 1 GW, far beyond the limits generally fixed by the power grid operators, even for special plants.
Experience from existing fusion experiments, fed by weak grids, reports about the use of local flywheel generators to store the majority of the energy necessary to provide the required pulsed power for the plasma formation, so as to reduce the demand from the grid.
In this paper, a new inductive energy storage scheme is studied: it can be applied both to tokamak or RFP experiments and can efficiently support the operation of the Central Solenoid (CS), without the need for resistive switching networks, thus with the advantage of energy dissipation avoidance The principle is applied to the case of a Fusion-Fission Hybrid Reactor, based on Reversed Field Pinch configuration operated exploiting the flux double swing.

The basic idea is to provide a SMES (Superconducting Magnetic Energy Storage) Coil (SC), pre-charged along with the CS one to the same current value. Then, for the ramp up of the plasma, the initial CS flux decrease is performed through the control of the voltage applied to the CS and recovering the magnetic energy on the SC, whose current increases. After CS zero crossing, the energy stored in the SC is transferred back to CS in such a way to provide the voltage for plasma sustainment during the flat top phase.
The paper will describe the concept and the scheme operation, which could be also considered for applications outside the fusion field; then, a conceptual design for the selected case is proposed.

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P4.092 The reactive power demand in DEMO: estimations and mitigation strategies

The European DEMO, which will follow ITER according to the Roadmap of the European fusion program, is presently under Pre-Conceptual Design. The DEMO Plant Electrical System (PES) will include the Power Supply (PS) and the Electrical Power Generation systems. In all the present tokamaks, the main magnets, which constitute the major loads, are powered by thyristor rectifiers; in DEMO they should be rated for several tens of kA and about ten kV. One of the main drawback of thyristor rectifiers operation is the large reactive power consumption when low voltage values are required by the load, as occurs during the major part of the plasma pulse. To satisfy the limitations imposed by the National Grid Operator and to reduce the currents flowing in the Medium Voltage Distribution System, the reactive power has to be compensated. In ITER, this is realized with a huge Reactive Power Compensation and Harmonic Filtering system, rated for 750 MVAR in total and based on Thyristor Controlled Reactors and tuned filters. Beside its additional cost and area occupancy, the high dynamics required to satisfy the reactive power limit in transient conditions represents a further severe challenge. These problems would be much amplified in DEMO, where the power ratings of the base converters are about five times higher than those of ITER. This paper presents the analytical model developed to estimate the reactive power demand of DEMO and the results achieved, assuming the traditional solution based on thyristor converters and the usual reactive power mitigation strategies. Then, an innovative approach, based on Active Front End converters, is proposed. A possible topology is studied, the achievable performance is derived by means of numerical simulations, and the pros and cons of the two options are discussed.

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P4.101 Conceptual design of the power supply systems for the Divertor Tokamak Test facility

The Divertor Tokamak Test (DTT) facility will be the link between ITER and DEMO as regards the power exhaust in the divertor structure. In order to accomplish this main goal, the Italian DTT will be a full-working tokamak. According to the last requirements and the physical, engineering and economic constraints, the current DTT design has a major radius above 2 m and an aspect ratio of 0.65. The DTT coil power supplies are designed to achieve a toroidal field of 6 T and a 5.5 MA plasma in single-null configuration.
The Pulsed Power Electrical Network (PPEN) will be able to feed for 100 s: 6 independent superconducting modules of the Central Solenoid (CS); 6 Poloidal Field (PF) superconducting coils; some internal coils for fast plasma control; the radiofrequency additional heating systems (ECRH and ICRH). The peak power absorption for the superconducting coils and the additional heating can reach 60 MVA and 160 MVA, respectively. A future PPEN upgrade is foreseen for the installation of a Neutral Beam Injector able to deliver up to 25 MW to the plasma.
The power supplies feeding the CS and PS coils are equipped with 4-quadrant converters and mech-

anisms for breakdown and emergency discharge. Current studies are devoted to the feasibility of ultracapacitor-based solutions for the design of alternative energy storage systems and assisting breakdown systems.
The total peak power absorption of the auxiliary systems fed by the Steady State Electrical Network (SSEN) was estimated around 50 MVA. This include about 3 MVA for the 18 Toroidal Field (TF) superconducting coils. In order to ensure a safety margin and to give some room to future upgrades, the request for the grid operator has been doubled. Several solutions are foreseen to keep the global power factor of the facility above 0.9.

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P4.111 High heat flux qualification tests of HELCZA facility

The high heat flux test facility HELCZA move forward in commissioning phase to the high heat flux qualification tests. The purpose of the qualification tests is the demonstration of facility's readiness for the testing of plasma facing components primarily first wall panels at high heat loads. For the qualification tests the flat cooper FW mock-up with representative size for electron beam irradiated windows of all types of FW panels was used. The heat loads coming up to 3.25 MW/m² (absorbed in cooling water) and cycling with at least 600 cycles (30 s beam on, 30 s dwell) was tested on this mock-up. The heat load testing was performed at inlet pressure 4 MPa, inlet temperature 70 ℃ and velocity of cooling water in cooling channels 11.0 m/s which are the parameters for real First Wall component. The qualification tests on this mock-up prove the facility full functionality to carry out the First Wall prototype tests and further for First Wall series.

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P4.103 Experimental study on anti-fatigue performance for TIG Welding and EB welding of RAFM steel plate

The Reduced Activation Martensitic/Ferritic steel (RAFM steel) is known as one of the most important materials for future fusion reactor blanket and the welding process is unavoidable for blanket structure, so it is extremely necessary to research his welding performance. Some simulations and experiments about anti-fatigue of RAFM steel welding specimens have been done in this paper. The anti-fatigue simulation of tungsten inert gas (TIG) welding and electron beam (EB) welding of RAFM steel was carried out by using ANSYS. The anti-fatigue performance for two types RAFM steel specimens of TIG welding and EB welding were also tested by SDS200 electro-hydraulic servo fatigue testing machine. The anti-fatigue of the weld joint both on TIG welding and EB welding was tested by applying the same gradient load. The experimental results were studied to analyze the impact of fatigue resistance for different welding forms of RAFM steel. It can provide some beneficial reference for the blanket design of china fusion engineering testing reactor (CFETR) in the future.

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P4.105 Studies of advanced divertor concepts and engineering design issues for a DEMO reactor

For a recent Japanese (JA) DEMO reactor design (Rp: 8 m size), the exhausted power to the SOL (Psep) is expected to be 200-300 MW, where large power handling (Psep/Rp = 24-35 MW/m) is re-

quired in the SOL and divertor. The conventional divertor design with the divertor leg length of 1.7 m was proposed, where the peak heat load at the divertor target was simulated to be less than 10 MW/m2 with the large radiation power fraction (Prad/Psep~80% level), and appropriate remote maintenance has been developed. Some advanced magnetic configurations are attractive for the divertor performance to reduce qtarget with smaller impurity seeding and Prad, which may reduce the divertor size. On the other hand, arrangement of the poloidal field coils (PFCs) and their currents are critical issue for the engineering design, pointed out in the previous JA DEMO design (Rp: 6m size). In this paper, advanced magnetic configurations such as snow flake divertor (SFD) and short super-X divertor (SXD) were investigated appropriate for the recent JA DEMO with decreasing the divertor size. The arrangement of PFCs was investigated, considering the maintenance port for the divertor, and minimal number of the divertor coils were installed inside the toroidal field coil (TFC), i.e. interlink-winding (interlink). Two and three interlink divertor coils were necessary to produce the short-SXD and SFD, respectively, and the maximum coil current for the SFD (~40 MA) was larger than those of the short-SXD (12-16 MA) since the SFD null was produced by merging two magnetic nulls. Issues of the technology and engineering such as R&D on the superconducting coil (reducing AC loss), interlink winding (so call "React and Wind") and reducing the electro-magnetic force on the interlink coil, are described. In addition, simulation result of the divertor performance (plasma detachment) will be presented.

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P4.106 Dynamics evaluation of hydrogen isotope behavior in tungsten simulating damage distribution at operating condition

Due to the higher melting point and lower sputtering yield, tungsten (W) is considered as the candidate for plasma facing materials (PFMs) in the future fusion reactors. Under working condition, W will be exposed to 14 MeV neutrons produced by D-T fusion reaction, as well as energetic particles such as hydrogen isotope, helium ion. The damages introduced by charge-exchanged particles are concentrated near the surface region, while that induced by 14 MeV neutrons are extended throughout the bulk. For the development of the effective fuel recycling and the safety operation, it is necessary to clarify the hydrogen isotope retention behavior in W with the consideration of damage distribution. In this study, irradiation damage distributions were controlled by 0.8 MeV and 6 MeV Fe ion irradiation with the damage concentration of 0.03-0.3 dpa (displacement per atom). According to calculation using SRIM code, the implantation depth of 0.8 MeV and 6 MeV Fe ions are about 0.5 μm and 1.5 μm, respectively. After the damage introduction by Fe ions, 1.0 keV deuterium ion (D2+) implantation was performed with the flux of 1.0 × 1018 D+ m−2 s−1 and up to the fluence of 1.0 × 1022 D+ m−2. The D retention behavior was evaluated by thermal desorption spectroscopy (TDS) at up to 1173 K with the heating rate of 0.5 K s−1. The experimental results showed that the total D retentions in the samples were decreased by the increasing the damage concentration introduced near the surface region by 0.8 MeV Fe ions. Retention of D trapped by vacancy clusters and voids, which are the trapping sites with higher trapping energies, was reduced, suggesting that the recombination of D atom into D2 on the W surface was enhanced due to D accumulation near the surface.

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P4.107 Additive manufacturing for realising a tailored tungsten-copper composite plasma-facing component heat sink

Tungsten (W) is a refractory metal foreseen as plasma-facing material in future magnetic confinement fusion devices. Furthermore, W containing metallic composite materials, such as W particle- or fibre-reinforced composites, are currently regarded as promising advanced materials to enhance

the performance and integrity of highly heat loaded plasma-facing components (PFCs). In principle, W is an intrinsically hard and brittle metal which means that the processing of W is rather difficult. That is also the reason why only rather simple geometries, e.g. flat tiles or monoblocks, are typically used for W armour parts in PFCs. Against this background, additive manufacturing (AM) could be a highly versatile approach for the realisation of W parts for PFCs. The characteristic feature of AM processes is that three-dimensional objects are created by sequential layerwise deposition of material under computer control, which implies that such a technology is capable of producing objects with more or less arbitrary shape. Within previous work, it was found that bulk pure W can be consolidated directly by means of powder bed based selective laser beam melting (LBM) with a rather high relative mass density (> 98%). The present contribution will summarise recent progress regarding the AM of pure W by means of selective LBM for material fabricated with preheated W substrates (up to 1000℃) in order to minimise the formation of crack defects within the deposited material. Additionally, it will be shown how the possibilities of AM technologies can be exploited for realising tailored W structures that can further be processed to structures with potentially superior layout compared to classical reinforced composite materials. Finally, results regarding the manufacturing and high heat flux testing of a PFC mock-up that makes use of such a tailored tungsten-copper composite material heat sink will be presented.

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P4.108 Development of 30 kW liquid metal handling system for plasma-facing experiments at DIFFER

The use of a liquid metal in the plasma-facing divertor components of the DEMO project has attractive properties such as self-healing and neutron resilience and a liquid metal handling system is considered as a candidate technology. The linear plasma device Magnum-PSI is capable of producing DEMO-relevant plasma fluxes which well replicate expected divertor conditions. The present research is focused on creating a prototype to demonstrate the feasibility of the liquid metal circulation through a plasma loaded target under a relevant plasma flux at Magnum-PSI. The system uses liquid tin as a cooling agent. The Sn pumping technology for this has recently been developed by Ushio and BLV Licht for a repetitively pulsed Sn-plasma based 13.5 nm Extreme UV (EUV) source. It is currently employed by TNO in the EBL2 nanolithography research facility where the liquid tin flow removes the 10 kW heat flux from the electrodes of the EUV source. For plasma facing experiments in a linear plasma device a prototype is proposed consisting of an electromagnetic liquid metal pump with a water-vapor cooled heat exchanger dissipating up to 30 kW. The system further consists of a heated pipeline to transmit liquid tin through the vacuum and open air parts of the setup as well as through the plasma loaded target installed in the DIFFER Magnum-PSI system. System design and operation requirements are analyzed and discussed based on the existing experience of the liquid tin based EUV sources and on the requirements to be met for the Magnum-PSI machine.

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P4.110 Hydrogen and deuterium behavior on gas driven permeation through tungsten deposition layer on nickel plate

Understanding of hydrogen isotope behaviors in plasma-facing wall is important from viewpoints of fuel control and tritium safety. Tungsten (W) is a candidate material of plasma-facing components. Although a sputtering rate of W is low, a certain amount of W deposition layer would be formed on the plasma-facing wall during a long time operation of a fusion reactor. However, hydrogen isotope behavior in W deposition layer has not been sufficiently investigated to date. In our previous work, hydrogen gas driven permeation through W deposition layer formed on nickel (Ni) plate was investigated and it was found that hydrogen permeation rate was larger than that in W bulk.

In this work, H2/Ar or D2/Ar gas at atmospheric pressure was supplied to one side of W deposition layer with Ni plate and the behaviors of hydrogen and deuterium released to another side were observed by a quadrupole mass spectrometer (QMS). The samples of W deposition layer were formed on circular plates of Ni by hydrogen RF plasma sputtering. The thickness of Ni plate was 20 micro meter and that of W deposition layer was 0.7 micro meter. The experimental temperatures were 180 – 500oC. The permeation curves by QMS were analyzed by TMAP code.
It was found that the permeation rate of deuterium was smaller than that of hydrogen but diffusivity of deuterium was not different from that of hydrogen. This suggests that deuterium solubility was smaller than hydrogen solubility. When deuterium gas was supplied to the sample after hydrogen permeation experiment was finished, not only deuterium but also hydrogen were quickly released. This hydrogen was considered to be released by an isotope exchange reaction between deuterium penetrating and hydrogen retained in the sample. These results give useful information for development of fuel recycling model in plasma-facing surfaces.

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P4.112 Measures to overcome deep crack issues in the tungsten tiles of the ASDEX Upgrade divertor

A massive tungsten divertor, Div-III, was installed into ASDEX Upgrade (AUG) in 2014. Div-III is an adiabatically loaded component and consists of massive tungsten tiles clamped into their supporting structures. Before installing the new component, extensive studies, including Finite Element Modeling and high heat flux tests in the test facility GLADIS, were carried out. After the first experimental campaigns the tile inspection reveals most of the tiles were cracked [1]. The difference between the high heat flux tests and the AUG behavior was attributed to mechanical loads due to disruptions and/or the thermal load profile and history. The actions to understand the cracks comprise tests with the Divertor manipulator, DIM-II, and FEM analysis of different target design options. DIM-II was used to test 'split' tiles, i.e. the deep crack is avoided by cutting a wide tile into two small tiles. FEM calculations were done to investigate the behavior of castellated targets with reduced tensile stress on top of the target and a clamping with a more elastic material, titanium instead of stainless steel. In addition more ductile tungsten heavy alloy (THA) was qualified for use in AUG. Based on this, a new set-up of tungsten tiles was installed in 2016. The paper reports on the inspection of this target arrangement after about 1000 plasma shots.

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P4.115 Depth profiles of deuterium in W coatings from the JET divertor measured by Glow Discharge Optical Emission Spectrometry

The GDOES (Glow Discharge Optical Emission Spectrometry) technique has been already used for investigation of erosion/re-deposition processes of W coated CFC tiles from JET. The challenge for GDOES was to measure the depth profile of deuterium together with the other elements from the W coatings exposed to JET plasmas. A tentative investigation using GDOES for deuterium analysis was performed by other authors, but the D depth profile was only qualitatively obtained.
Since reference samples for D calibration are not available on the market, these samples have been produced in our laboratory by High Power Impulse Magnetron Sputtering (HiPIMS) and Combined Magnetron Sputtering and Ion Implantation (CMSII). The D concentrations were measured by NRA (Nuclear Reaction Analysis) and ToF ERDA (Time of Flight Elastic Recoil Detection Analysis). Then these samples were used to calibrate the D channel of the GDOES machine.
Special samples cored from Tiles 1, 6 and 3, 7, 8 exposed in JET in the periods 2012-2014 and 2011-2014 respectively, were analyzed by GDOES and SEM. The following aspects can be emphasised:

- Be has contaminated both inner and outer divertor tiles.
- The thickness of the Be layer varies from about 2 microns on Tile 8 to about 44 microns on Tile 3.
- D penetrates into the coating, but its profile does not seem to follow that of Be. On Tile 3 where the Be deposition is very strong, the deuterium concentration is the lowest (0.2-0.6 at.%). On Tile 8 with the lowest Be deposition, the D concentration is highest (3.2 at.%).
- Small peaks of D have been detected at the interfaces.
- It is important to notice that the concentration of D found by GDOES is in very good agreement with that found by NRA (1 – 3.5 at.%).

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P4.118 Reactor studies of hydrogen isotopes interaction with lithium CPS using dynamic sorption technique

At present, the study of structural and functional materials' properties of fusion reactors is carried out as a part of implementation of ITER and DEMO projects.
The research of liquid metals application possibility as a plasma facing material (PFM) is one of the most important area. Studies carried out on this subject have shown that lithium is a good material for use as a PFM in a fusion reactor. The use of liquid lithium is more attractive, if lithium is enclosed in a capillary-porous system (CPS). Compared to solid PFMs lithium CPS have several advantages: resistance to properties degradation, ability to surface self-repair due to capillary forces both in plasma discharge conditions, plasma breakdowns and in the case of ELMs. An important problem in the "plasma-wall" studies is the investigation of the processes of hydrogen isotopes interaction with lithium CPS under neutron irradiation.
This paper presents the results of hydrogen isotopes interaction with lithium CPS under reactor irradiation at temperatures of the CPS sample from 200 to 800 ℃. A special ampoule device with a tubular sample of lithium CPS was developed for experiments. The IVG.1M reactor (Kurchatov) was used as a source of radiation. The experiments were carried out by the method of dynamic sorption: a constant stream of deuterium was swept through the tube sample with constant degassing and fixation of gas composition in the experimental ampoule device.
The time dependences of gas composition change in the ampoule device during irradiation were recorded at different sample temperatures and deuterium fluxes. The results obtained were compared with non-reactor experiments. The mechanisms of deuterium sorption on the surface of liquid lithium were determined, and the level and composition of the tritium-containing molecules released from lithium were estimated.
The work was performed within the framework of the ISTC #K-2204 project.

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P4.123 DEMO First Wall misalignment study

Within the DEMO first wall 3-D shape design activity the first study of the effect of misalignment has started. Such assessments have been conducted in the past for ITER and penalty factor maps have been created; this route could be a feasible approach in the case of DEMO wall design also. This paper details the first set of computational tests that have been carried out for DEMO. The test cases focus on the steady-state plasma operation (flat top). The aim is to understand the effect of basic misaligned cases, for example, radial protrusion/recession or poloidal rotation of a single module. To do so 3-D particle tracing software codes such as SMARDDA and PFCflux have been used to create heat flux maps that reach the first wall surfaces. The obtained heat flux maps combined with the specified radiative heat load are used as input for simplified FE models of the blanket modules. As a result, not only the effect of misalignment on heat flux, but also on the temperature and stress distribution can be estimated.

The paper describes the test matrix that has been studied and shows how the obtained results can be implemented in ANSYS in the identified critical cases. The results obtained from the nominal heat flux map are compared to the misaligned cases. The peak temperature mitigating effect of 3-D heat conduction is discussed. This work paves the way to assess more realistic combined misaligned cases (such as misalignment from different thermal expansion, or due to electromagnetic loads etc. of neighbouring blankets) in the future. Differences of single-module and multi-module segments in terms of misalignment is also discussed.

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P4.124 High heat flux test results for a thermal break DEMO divertor target and subsequent design and manufacture development

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The European roadmap to fusion electricity attributes critical importance to the development of reliable plasma-facing component (PFC) technology. In the EUROfusion divertor project, a range of PFC concepts are explored, and two "Phases" of small-scale mock-up design, manufacture and high heat flux testing are underway. The rationale is to understand shortcomings revealed by the first phase in order to improve designs and manufacturing methods for the second phase. In this paper, we focus on the Thermal Break concept, which is an evolution of the ITER tungsten/CuCrZr monoblock design in which the copper interlayer has geometric features to reduce conductivity and stiffness thereby alleviating stress in the PFC. In Phase 1, six small-scale mock-ups of this design were subjected to high heat flux testing, with applied heat flux up to 25 MW/m^2 and thermal cycling of up to 500 cycles at 20 MW/m^2. Although all six mock-ups survived the campaign and maintained 20 MW/m^2 heat exhaust capability, at least two show signs of progressive damage. Detailed examination of these mock-ups was carried out to understand the damage mechanisms, using ultrasonic imaging, infrared thermography (SATIR), and destructive microscopy. Although there are signs of tungsten surface cracking, the predominant damage mode is not by "deep cracking" of monoblocks, but substantial permanent deformation of the interlayer features.
The results of the first phase manufacture and testing have informed the design and production of the second phase mock-ups. The manufacturing procedure has been updated to eliminate the need for one of the vacuum brazed joints, and the interlayer grooves have stress-relieving radii which are found by numerical analyses to significantly reduce the interlayer plastic strain range. Interlayer design parameters were selected following use of response surface-based design optimisation. Mock-ups of the new Phase 2 design are expected to be high heat flux tested during 2018.

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P4.127 Development of self-passivating W alloy / Eurofer brazed joints

The development of the future fusion reactor is highly associated to the improvement of the current materials, specially tungsten materials, to withstand the extreme conditions giving inside the reactor vessel during service life. Tungsten has extraordinary physical characteristics as plasma facing materials (high thermal conductivity, sputtering resistance and melting point). However, it has also two main disadvantages: 1) its high brittleness and 2) its high corrosion rate at high temperatures in an oxygen atmosphere. In order to solve the second inconvenient self-passivating materials based on the addition of Cr and Y2O3 particles have been developed to enhance corrosion behavior of the material. These new materials have to be joined to other materials (i.e. Eurofer) in order to conform

the plasma facing components. Therefore, joining technologies need to be implemented to meet the requirements of the reactor environment.

The present work proposes a brazing procedure to join a self-passivating tungsten alloy (W-10Cr-0.5Y2O3) with Eurofer. The results indicated the achievement of high quality W-Eurofer joints using 80Cu-20Ti filler material. The resultant microstructure of the braze changed considerably compared to the brazing attempts with pure tungsten, which is associated to the reactive character of chromium. A high interaction between molten filler and W base material has been detected, with preferential grain boundary penetration of Cu-Ti into W alloy.

The mechanical properties of the joints were evaluated by means of microhardness and shear strength tests. The microhardness study of the self-passivating tungsten alloy indicated that the mechanical properties and, therefore, the microstructure of the base material have not been affected by the brazing process because their hardness values measured in the as received conditions have not been modified. Regarding to the strength of the joints, a shear strength of ~ 90 MPa has been obtained, which ensure the operative brazeability and the metallurgical brazeability.

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P4.129 Investigation of beryllium damage under transient plasma heat loads in wide range of temperatures and heat loads

The first wall panels of the ITER main chamber will be completely armored with beryllium. The primary reasons for the selection of beryllium as an armor material for the ITER first wall are its low Z, high oxygen gettering characteristics and also high thermal conductivity. During plasma operation in the ITER, beryllium besides low cyclic heat loads (normal events) will be suffered by high transient heat loads, such as ELMs, disruptions, VDE, etc. (off normal events). These transient loads cause rapid heating of beryllium surface and can result in significant changes in surface and near-surface regions, such as material loss, melting, cracking, evaporation and formation of beryllium dust as well as hydrogen isotopes retention both in the armor and in the dust. It is expected that the erosion of beryllium under transient plasma loads such as ELMs and disruptions will have significant impact on lifetime of the ITER first wall.
This paper presents the main results of numerous experiments carried out during some last years at QSPA-Be plasma accelerator in Bochvar Institute. QSPA-Be plasma accelerator is capable to provide plasma (hydrogen or deuterium) and radiation heat loads on target surface relevant to ITER ELMs and mitigated disruptions. Special Be and Be/CuCrZr mock-ups were tested by hydrogen/deuterium plasma streams (5 cm in diameter) with pulse duration of 0.5 ms in a heat loads range of 0.2-2.2 MJ/m2 and maximum quantities of plasma pulses up to 100-250 shots. The angle between plasma stream direction and mock-ups surface was 30o. During the experiments, the mock-ups temperature has been maintained in the range of RT-500℃. Two beryllium ITER grades: TGP-56FW (RF, Bochvar Institute) and S-65C (USA, Materion Brush) were studied in these experiments. Influences of plasma heat loads, surface temperature and quantities of plasma pulses on the Be erosion and surface damage are presented.

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P4.133 Study of the ports effects on the CFETR radiation shielding

China Fusion Engineering Test Reactor (CFETR) is a tokamak reactor under design. Due to the maintenance, assembly of the blanket and heating, diagnostics of the plasma, ports must be opened on the vacuum vessel and the blanket module of CFETR. However, the ports affects the radiation shielding performance of CFETR, especially the equatorial ports where the neutron wall loading

appears a peaking value. In this paper, the study focuses on the effects of ports on the CFETR radiation shielding performance.
The analyses have been performed with the Monte Carlo code MCNP and nuclear cross-section data from the FENDL-3.1b data library to estimate the radiation load of the various components around the ports. The three-dimensional MCNP model of CFETR was generated from the reference CAD model with the McCad geometry conversion tool. The model includes blanket modules, shield, divertor, vacuum vessel, upper port, equatorial port, lower port, thermal shield, TF coils, PF coils, CS and cryostat. The neutron flux densities, the nuclear power density, the displacement damage rate, the helium production rate of the vacuum vessel and thermal shield around the ports and the radiation load on the toroidal field superconducting magnet coils behind the ports were calculated. The calculation shows a significant increase of the radiation load around the ports. The detail results will be presented in this conference.
Keywords: CFETR, Port, Radiation Shielding

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P4.134 Numerical design approaval of port liner and bellows protection

The world's largest modular stellarator, Wendelstein 7-X (W7-X), is in operation since 2015. Stepwise increase of operation parameters has its final goal in the demonstration of steady-state operation capabilities with pulses up to 30 minutes. Such pulses require a constant heating of the plasma due to losses by plasma-wall interactions, particle drift and radiation losses. The latter are assumed to occur with resulting loads of up to 5 kW·m-2 in average on the surfaces of the ports. Thus, a thermal shielding is required for the port walls, in order to keep the average wall temperature below 80 ℃. Otherwise, unsustainable heat radiation loads on the superconducting magnetic coil system of W7-X could emerge and cause a quench incident.
Due to gaps between port and first wall components, like graphite tiles and panels, the weld directly faces a small fraction of the plasma radiation, which subsequently requires protection. In the inner cavities of the ports, short length microwave stray radiation (ECRH) is the dominating radiation component. This renders usual protection measures, like shields throwing a shadow, for the thin walled bellows useless.
The paper describes the calculation of the relevant heat loads, based on a 1-way ray-tracing code [1] and the developed radiation-shielding concept for the port. The concept consists of a water-cooled port-liner, a passively cooled sheet metal gap-closure, attached to the liner and designed to protect the exposed welds and a copper bellows protection, blocking ECRH stray-radiation from sensitive bellows. Finally, the vital ability to serve as a steady-state capable shielding concept is demonstrated here.

[1] S. Bozhenkov, https://bitbucket.org/sboz/meshlib

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P4.135 A sophisticated remote maintenance method for the target assembly of Advanced Fusion Neutron Source

Remote maintenance (RM) of highly activated components in the test cell (TC) is a key issue of design of accelerator-driven (IFMIF-type) fusion neutron sources. We newly propose a sophisticated RM method for the target assembly (TA) of an IFMIF-type Advanced Fusion Neutron Source (A-FNS), aiming at improving maintainability of the in-TC components in this study. Basic ideas of the proposed RM method are (i) to make the larger clearance of 50 mm between the TA and test module (TM) than that of the IFMIF/EVEDA design, 2 mm, and (ii) to reach out the RM manipulators to the TA from the beam dump side, not behind the TA backplate. We performed a kinematics

analysis with detailed CAD data of the TA and RM tools. The analysis results indicate that the TA can be removed/installed by the newly proposed RM method without removal of the TM, and the TA RM can be independent of any TM RM schemes. The newly proposed RM method can drastically improve the maintainability of the in-TC components and the availability of neutron irradiation. A neutronics analysis has been performed to check the compatibility of the newly proposed RH method with the TM nuclear properties. The analysis indicates that the larger clearance brings decrease in the gradient of the neutron irradiation damage rate (dpa/fpy) at the material test specimen in the TM, which is beneficial for post-irradiation experiments of the irradiated specimen. Although the dpa/fpy is decreased by the larger clearance, but it can fully satisfy the requirements for the neutron irradiation damage for evaluation of material mechanical properties and DEMO design. We conclude that the newly proposed TA RM method is technically feasible and also compatible with the neutron irradiation tests.

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P4.136 New methodology to improve execution efficiency and safety in remote maintenance tasks

Remote Handling (RH) maintenance in hazardous environments is a challenging task to be performed on systems and components to ensure that they work as per design and that the requirements on the plant availability are fulfilled. According to this, systems and components have to be designed for easy assembly. This approach leads to an improvement and efficient maintenance process, reducing assembly and maintenance time and costs. Over the time, guidelines and methodologies have been developed in order to support design, assembly and manufacturing stages such as Assembly Evaluation Method (AEM) or Design For Assembly method (DFA). However, these methodologies cannot cover all the design aspects to be implemented for the remote maintenance of systems and components.
In this article, a proposal for a new methodology is presented with the aim of improving the execution, efficiency and safety of remote maintenance tasks. The methodology is composed of a set of nine rules, which are divided into specific blocks with the aim of simplifying the current complex procedures. These blocks are related to: design of systems and RH equipment, logistics, development of maintenance procedures, and simulation of RH tasks. This methodology relies on the initial cooperation between the different working teams involved in the components life cycle. This approach leads to an iterative process that allows the optimization, design and proposal of improvements. The result is a speed up of the process and more robust concept of maintenance.
Thereby, the application of this new methodology has derived in a simplification of the procedures, less system failures, and saving of costs and time. Considering that the method provides specific work sequences and control, it promotes collaboration and integration of different areas of expertise associated with the tasks.

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P4.139 Development and Validation of Cryostat Finite Element Model with Unique FE Method

The ITER Cryostat, the largest SS vacuum pressure chamber ever built which provides the vacuum confinement to components operating in ITER ranging from 4.5 k to 80 k. Cryostat Design Model was qualified by ITER. As a Safety Important Class system, Design qualification at every change in its development and installation phases is mandatory. The Cryostat system is currently at manufacturing stage, several Deviation request are being reported e.g. tolerances change. These changes affect the behavior of Cryostat which needs re-assessment. The conventional design approach in Finite Element method (FEM) needs significant time and effort, as incorporation of changes calls for

redevelopment of full mathematical model (MM).
In this paper a "unique method" of developing FE model for complex systems like Cryostat is presented, which typically addresses above need and the method is qualified with results of Cryostat engineering model (CEM). This unique method involves dividing and meshing of the big components in to sub components, so the full Cryostat is divided into 30 sub components and MM of these individual components are developed. These sub components are integrated using suitable constraint equations to create full FE model. Then the integrity of model is assessed using the modes shapes. This unique method enables to incorporate component level changes without affecting the full FE model, thus saving time and efforts of re-development of MM.
For qualification of developed FE model category II loading and selected load combinations are applied. The results obtained are in close approximation with CEM results. As the present need was to address the changes of MM, so further Cryostat manufacturing FE model (CMM) is developed with this unique approach. It is analyzed for category II loading and selected load combination.
This paper gives detailed insight about the developing and qualification of the Unique Method and analysis results.

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P4.141 Deformation Modeling and Compensation Control of Manipulators for DEMO

It is foreseen the most challenging tasks currently in remote maintenance of DEMO are the remote handlings of multi module blanket segment and divertor. Both of them are large and heavy, and must manoeuvre through precise trajectories in a limited space by remote manipulators. The manipulator deformation, due to the heavy payload, will deviate the handled component from the desired trajectories, which may cause the collisions of the manipulator with the surroundings. In order to implement the remote handling processes accurately by highly automatic manipulators, the manipulator deformation model has to be developed to predict its deformation displacement. Thus resultant displacement with respect to the desired trajectories can be compensated in control system in real time.
The hybrid deformation modeling method for generic serial manipulators has been employed, and the compensation method in a feedforward control system has been developed. Artificial neural networks (ANNs) are used to model equivalent deformation physics of the consisting links and joints, and the hybrid deformation model is constructed by integrating the ANNs deformation into kinematics. The distal deformation displacement of manipulator end-effector are computed through hybrid deformation model, and taken as extra trajectory errors for the compensation in control system. The errors are fed forward to a deformation compensation controller, which inversely compute the compensations in joint space. The joint's compensations, together with the joint interpolation errors are fed to a PID controller, which finally outputs the commands for the joint driving system.
The proposed deformation modeling and compensation methods are applied to a serial manipulator. The comparison results of trajectory tracking are presented, with respect to the control system without feedforward deformation compensation controller (FDCC). The results indicate the trajectory tracking accuracy can be significantly improved with the FDCC, and the methodologies developed herein can be extrapolated to realistic manipulators in DEMO.

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P4.142 Heat Management for Water-Hydraulic Systems at ITER Remote Handling

The full exchange of the ITER divertor is performed with the Divertor Remote Handling System during ITER long-term maintenance campaigns. Access to the vessel is possible through the lower

maintenance port. The size of the port is highly constrained, therefore, high power density actuation systems are required to lift and transport the 10-tonne cassette assemblies in and out the vessel. It has been found that water-hydraulics can deliver the required power and tracking accuracy.
The ambient temperature within the divertor area is around 50℃, which tends to be at the upper limit of allowed temperature range for water hydraulic components. Cooling hydraulic fluid below this with traditional methods is practically impossible. In this study, heat management of two water-hydraulic powered systems were considered, namely Cassette Toroidal Mover (CTM) and Cassette Multifunctional Mover (CMM). Hydraulic power for the CTM cannot be externally supplied as the umbilical connecting CTM to the outer world is packed with electric wires for motors and control as is. Therefore, the Hydraulic Power Unit (HPU) needs to be situated within the CTM chassis. Power for the CMM hydraulics is supplied from the transfer cask used for transporting plant components and maintenance equipment between divertor area and the hot cell. It is practically the only place the HPU can be housed in. Hence, a project to develop and analyse heat management methods for the Movers was undertaken.
Multiple heat management methods were devised and analysed. Their pros and cons were weighed against each other and the best solutions for these particular cases were selected after trade-off analyses. As a result, rather a traditional fluid/air heat management solution was selected for the CMM and a solution which includes no dedicated cooling was selected for the CTM. Whether the solutions suggested herein are applicable to other ITER systems is discussed.

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P4.143 Water quality and cleanliness effects on water hydraulic components in ITER remote handling maintenance systems

Remote Handling (RH) equipment are deployed to exchange the ITER Divertor, segmented in 54 cassettes. The RH equipment are powered and controlled with water hydraulics, using self-supplied demineralised water as a pressure medium. Water hydraulics servo control technology has been successfully proven at Divertor Test Platform 2 (DTP2) with a full-scale prototype, namely Cassette Multifunctional Mover (CMM). However, the tests at the DTP2 have been carried out at 20℃ ambient temperature while the maximum in-vessel ambient temperature during long-term maintenance operations can reach up to 50℃. Demineralised water at high temperature is known to be chemically aggressive, leading to potential problems such as corrosion and seal material degradation. This could result in accelerated hydraulic component degradation and in water particle contamination. The latter is especially worrying as water hydraulic servo valves require high levels of water cleanliness in order to function correctly.
A test was commenced to ascertain the impact of demineralized water at high temperature on the selected water hydraulic technology in terms of performance and component deterioration. A heated chamber was constructed to emulate the 50℃ ambient temperature of the divertor area. The water hydraulic system test bed inside the heated chamber was a loaded single-joint system with the hydraulic components of the CMM design. The test system ran for 2000 hours. Water impurities were monitored regularly over the testing period.
The test results showed that the filtration system could keep the water cleanliness up to the required level. Deterioration and corrosion was found in the post-test disassembly of the components, even though the component materials were specified as stainless steel and other non-corroding materials. A discussion on component material compatibility and effect of heat treatments, water quality and cleanliness, and management thereof, based on the results and observations on the test period is included in this paper.

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P4.146 Preliminary source terms investigation for decontamination of fusion reactor in-vessel component maintenance process

in hot cell

The fusion reactor produces a large amount of tritium dust in the vacuum vessel due to the plasma unstable events, decontamination is an indispensability dispose in the IVC maintenance and decommissioning, it also plays an extremely important role in reducing the radioactive pollutants diffusion, cumulative radiation dose and staff occupational exposure level, controlling radioactive effluent of nuclear facilities and enhancing remote handling equipment life and reliability and even series environmental protection problems.
Evaluating of hot cell interior decontamination solution for tritium dust is a complexity task. The source terms investigation, as an important basis for the selection of cleaning methods and the formulation of the decontamination process, is the foundation for the selection of clean and decontamination methods and the development of remote handling dedicated equipment. This paper is based on fusion reactor hot cell indoor cleaning decontamination process of latent source items, combining comprehensive consideration with dust composition of mainstream fusion device data datum and representations after future D-T burning. Collecting, analysis and concluding the characterization, composition, morphology, status, potential risk of dust in vacuum vessel, especially the impact of tritium retention for the decontamination process. Combining consider the confinement of the hot cell facility, release relevant evaluation criteria for the selection of protective measures and decontamination solution.

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P4.179 Separation and recycle of Plasma Enhancement Gases using a CMS membrane

The research focuses on testing carbon molecular sieve membrane (CMS) for separation of the exhaust gas from fusion reactors.
The exhaust gas in tokamak demonstration reactor (DEMO) consists of more than 90% of unburned fuel gas (D and T) and the remaining part will be He, plasma enhancement gases (PEGs) and impurities. Plasma enhancement gases (PEGs) (such as: nitrogen, neon, argon and other inert gases) are injected into the plasma in order to reduce the power load over the plasma facing component. In demonstration reactor (DEMO) is foreseen to recover the fuel gas (D and T) and PEGs.
The CMS membranes have been supplied by Media and Process Technology Inc. (Pittsburgh, PA, USA).

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P4.149 Electromagnetic and structural analysis of COMPASS Upgrade vacuum vessel during disruptions

COMPASS Upgrade tokamak is a medium-size high-magnetic-field device currently in the conceptual design phase [Panek et al. Fus. Eng. Des. 123 (2017) 11-16]. Due to the high plasma current (up to 2 MA) and the strong magnetic field (up to 5 T), large electromagnetic forces on conducting structures surrounding plasma are expected during disruptions. To address this issue, electromagnetic loads on the vacuum vessel during vertical displacement events, thermal and current quenches are calculated by means of ANSYS numerical simulations. The numerical results are compared with analytical estimates. The effects of in-vessel passive stabilizing plates and their electrical resistivity on the force distribution in the vacuum vessel are considered. Different access-port geometries and their influence on the vacuum vessel mechanical strength are analysed. The study reveals optimal choice of the vacuum vessel parameters which guarantee safe operation of the COMPASS Upgrade tokamak.

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P4.150 Parametric Modelling of Remote Maintenance Strategies and Concepts for EU DEMO

One of the major challenge in the commercialisation of magnetic confinement fusion is maintaining the powerplant reactor in a sufficiently short period of time to achieve commercial levels of plant availability. To inform the development of an appropriate remote maintenance strategy for EU DEMO, a simplified, parametric model, called the Maintenance Duration Estimator (MDE), has been created to enable comparative analysis of various remote maintenance concepts and strategies. These models will be used to inform the development of remote maintenance strategies and methodologies for EU DEMO.

Using the MDE, a model of the duration of various maintenance activities was created, which in turn was used to develop a 20-calendar year maintenance program for EU DEMO. The currently proposed Ex-Vessel maintenance program was also modelled. Here we will present the conclusion of this work, which estimated the lifetime durations of in-vessel and ex-vessel maintenance activities. This found that 26% of DEMOs lifetime is dominated by ex-vessel maintenance activities, compared to 19% in-vessel operations, resulting in 41% powerplant uptime. Therefore, design considerations to support maintenance should be integrated into the plant design from the outset to minimize plant downtime

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P4.151 A simulation framework for dynamic analysis of flexible robotic mechanisms in DEMO remote maintenance

The numerical analysis of robotic mechanisms for remote maintenance and inspection inside nuclear fusion reactors has to face several issues. Indeed, these robots are subject to large deformations, which are either induced by their own mechanical structure or by the heavy payloads which they usually handle. In many applications, robotic systems are usually modeled with rigid elements, although different formulations have been developed over the years to handle link and joint flexibility. In order to limit the mathematical complexity, and thus the computational time required for solving the equations of motion, these techniques usually assume linear elasticity and small deflections. In nuclear fusion environments, in particular for the future DEMO reactor, the large loads induce nonlinear behaviors which need to be predicted with details so as to plan safe and collision-free autonomous remote operations. Therefore, an accurate nonlinear finite element approach based on geometrically exact models is highly desirable.
To simulate robots subject to large deformations, we use a screw-based nonlinear finite element formulation. Finite element procedures have the advantages of: (i) modeling manipulators with rigid and/or flexible arms; (ii) modeling manipulators with rigid and/or flexible joints; (iii) modeling manipulators structured in a serial or parallel topology using the same systematic approach. These features are particularly appealing in the context of robotic autonomous maintenance and inspection in tokamak machines, where high complex mechanisms are involved in usual operations. Furthermore, the screw-based approach yields to a global parametrization-free framework. This allows solving the equations of motion using geometric time integrators, which significantly speed up the computation.
In this work, we present a simulation framework for dynamic analysis of flexible robotic mechanisms used for nuclear fusion applications. We show the potentialities of the approach by simulating: (1) an hybrid serial/parallel flexible robotic mechanism; (2) an hyper-redundant flexible manipulator.

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P4.152 The DTT device: advances in conceptual design of first wall vessel and cryostat structures

In this work the authors present the latest progresses in the conceptual design of the first wall and the main containment structures of DTT device. The previous DTT baseline design was reviewed in terms of both materials and plasma shape. This in turn led to a new all-welded double-wall vacuum vessel structure, made of AISI 316L(N) stainless steel. While the basic design has still 18 sectors, ports configuration has been revised according to new design constraints about remote maintenance, diagnostics and heating equipment. Supports have been also designed for the first wall, which was conveniently segmented in view of assembly and remote maintenance. FEA analyses confirmed that mechanical structures well withstand all primary and secondary loads, in particular the ones resulting from possible plasma disruptions. The design principles of the cryostat were chiefly based on cost minimization and functionality; thus it was conceived as a single-wall cylindrical vessel supported by a steel frame structure. The same structure holds vacuum vessel and magnets.

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P4.154 Progress on reliability of remote maintenance concept for Japan's fusion DEMO reactor

The banana-shaped segment transport using all vertical maintenance ports (BSAV scheme) was selected as the primary maintenance option on Japanese DEMO. Among various engineering issues on the BSAV scheme, recent progress on remote maintenance (RM) focuses on in-vessel transferring mechanism of the segment, support structure of the segment and pipe connection. In the BSAV scheme, cooperative operation and complex attitude control consistent with required installation accuracy and support structure of segment by full-remote operation are important issues on the maintenance. Therefore, the basic concept required for the RH device design is a simplified transferring mechanism for the blanket segment, that is, no rotation in vessel. Previous lift system was a cable crane, which catches the segment at the top of the segment. However, when one simply unilaterally supports or hoists the banana segment at the top of the segment, the segment tends to swing because its center-of-mass of the segment is away from the vertical axis directly beneath the supporting point. Updated RH equipment has a telescopic manipulator with a guide flame and the end effector for suppressing the swing. The segment attitude is kept by the end effector, which is sustained by the rigidity of the telescopic manipulator. The RH equipment is composed of the end effectors (grippers) for the banana segment, a power manipulator, a telescopic guide and a carrier. These components allow a three-axis attitude control of the banana segment in the vertical (Z), radial (R) and toroidal (f) directions. The role of the end effectors is to grip the segment at the top and bottom and to suppress the swing. The attitude control of RH equipment is made with a lift in the Z-direction, a link mechanism in the R-direction and a bogie in the f-direction, respectively.

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P4.155 A design study of vacuum vessel and pumping system for COMPASS-U Tokamak

In Early 2018 a new challenging project of building unique Tokamak device Compass-U has begun. Apart from others, the technical attributes will include elevated operating temperatures of vacuum vessel and plasma facing components, active cooling of temperature sensitive diagnostics, cooling of the magnetic coils to LN temperatures and etc. The listed requirements of operating parameters pose high demands and difficulties on the design and performance of the whole vacuum system of the experimental chamber and its cryostat.
In the presented contribution we discuss the conceptual design of vacuum vessel and related vacuum

pumping system. Due to the elevated temperature of the vessel wall and plasma facing components, the integral outgassing rate of in-vessel components and the vessel itself had to be estimated. Preliminary calculations of the pump down curves was numerically calculated in order to achieve optimal operation parameters before discharge in reasonable times. These preliminary results are compared with respect to experiences achieved during operation of the existing device Compass.

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P4.158 Use of morphing techniques within weld distortion analysis to simulate the size customization of parts during assembly and welding of the Vacuum Vessel PS3 outer shell

During welding of large, massive and complex assemblies, such as the ITER Vacuum Vessel PS3, the accumulated shrinkage of parallel welds may cause a cumulative distortion effect which significantly varies the dimensions and geometry of the welded portion while still unfinished. In case of the PS3, as consequence of such distortion, the space left to fit the outer shell plates into to the welded portion is expected to differ from nominal dimension by several millimeters. The outer shell plates to be added and welded to complete the assembly are foreseen to be customized to the proper size after dimensional survey of the space to allocate them. Such customization will allow the bevel dimensions for the remaining welds to be nearly of nominal size and the weld parameters and procedures still to be valid.
In the field of welding distortion simulation with the use of the finite element birth and death technique, all parts of the assembly and bevels between them are included in the model from the beginning of the simulation with their nominal sizes. The process described of part customization needs to be taken into account in the simulation. Otherwise the distortion of the welded portion will deform the still dead elements of un-welded bevels in the model. This would be a relevant source of inexactitude for the analysis since such bevels would not have anymore the proper size, and dimensions of parts added to the assembly would also differ between simulation and the actually manufactured ones.
A methodology is presented to simulate, through morphing techniques, the size customization of the parts being added to fit into a distorted assembly, maintaining also the bevel sizes within proper dimensions. The results show the outcome of the investigation carried out for welding distortion analysis of PS3 using such methodology.
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P4.159 Deep feature based detection and location of the damaged first wall for EAST Tokamak

EAST Tokamak is a complex fusion device which requires high-quality first wall condition for the long pulse and H-mode plasma operation. During plasma phases, some of the first wall components are damaged due to high thermal stress and electromagnetic force and control is necessary in case of doubt about their condition. Detection and locating the damaged PFCs are the precondition to determine whether to continue the experiment or shut down the machine to maintain components. This paper presents a process to detect and locate the damaged first wall automatically based on the visual features learned by deep learning. Compared to classic object recognition and detection methods based on low-level geometric features, the proposed method is able to extract high-level features which are more robust to disturbances and accomplish higher recognition and detection

accuracy. The proposed method is validated in the simulation environment consisting of models of EAST Tokamak and EAMA (EAST articulated maintenance arm) robot.

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P4.160 Lithium-vaporization properties of Li8ZrO6 at medium temperature in 0.01% H2/He gas flow

Tritium breeders is required to have a high lithium atom density from the viewpoint of tritium breeding ratio. The first candidate material being studied in Japan at the present time is Li2+xTiO3 which is Li2TiO3 containing excess Lithium [1], however development of a material having even higher Lithium atom density is still under way. Li8ZrO6 has the highest lithium atom density except for Li2O which is a highly-reactive and unstable material. Although the lithium vaporization behavior from Li8ZrO6 at 900℃ which is the maximum temperature which is assumed for Li2TiO3 and Li2+xTiO3 has been reported [2], research at lower temperatures where most tritium breeders are exposed has not been implemented yet. To elucidate the possibility of Li8ZrO6 as a tritium breeder, it is necessary to clarify the stability of its sintered body under actual use environment. In this study, Li8ZrO6 powder is synthesized in partial oxygen pressure controlled atmosphere by solid state reaction [3], and the lithium vaporization behavior and thermochemical stability of the Li8ZrO6 sintered body in a representative environment (320-900oC, 0.1% H2/He) to which the tritium breeders are exposed, lithium vaporization behavior and thermochemical stability are evaluated. Lithium vaporization rate was measured by gravimetric method and inductively coupled plasma, and crystal-phase change was analyzed by x-ray diffraction. In the relatively low temperature range, the sintered body of Li8ZrO6 maintained higher lithium atom density than the sintered body of Li2TiO3 and Li2+xTiO3 even after two years assumed as the actual use period. However, when the carrier gas contained water vapor as an impurity, LiOH was easily produced.

[1] T. Hoshino, et al., Fusion Engineering and Design, 82 (15-24), 2269-2273 (2007).
[2] K. Shin-mura, et al., Fusion Engineering and Design, 124, 762-766 (2017).
[3] K. Shin-mura, et al., Fusion Engineering and Design, 109-111, 1739-1743 (2016).

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P4.161 Fabrication of thin-walled fusion blanket components like flow channel inserts by selective laser manufacturing

The development of new manufacturing methods for the production of key components for nuclear fusion reactors by selective laser manufacturing (SLM) is currently under investigation at Karlsruhe Institute of Technology. SLM offers great potential compared with conventional manufacturing methods, especially for fabrication of thin- and double-walled structures like sandwich-type flow channel inserts (FCIs) or components with complex internal geometries. Material qualification tests with samples produced by SLM Eurofer (identical chemical composition) have shown almost similar material behavior than conventionally fabricated components. Feasibility studies to demonstrate the producibility of sub-components for fusion reactors with complex shapes and structures by SLM technique are ongoing. In conjunction with these feasibility studies, complex 3D structures such as a thin- and double-walled FCI for dual-coolant lead lithium (DCLL) blankets with the 3D shape similar to blanket contours have been successfully manufactured and tested on a preliminary level. Due to limitation in maximum dimensions of components produced with present-day SLM machines, only a model-sized prototype of a sandwich-type FCI had been manufactured. For that reason, the present concept for full-scale applications relies on a modular arrangement assembled from several segments with ceramic connector parts. To overcome space limitations new approaches for bigger respectively longer parts are currently under development. The paper focuses on the principal feasibility of SLM technique to fabricate thin-walled FCIs. The complexity of the fabrication process is

outlined and application of SLM technique for production of other fusion blanket sub-components like thin-walled cooling plates with internal cooling channels is addressed.

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P4.163 Investigation of Alumina Films Formed over Aluminized RAFM steels by Plasma Assisted Heat Treatments

Reduced Activation Ferritic Martensitic (RAFM) steel is currently under intense consideration as a structural material for the blanket applications in fusion reactor. The concept of blanket module utilizes both solid i.e. Li2TiO3 and liquid breeder material, i.e. Eutectic Pb-17Li operating at 320-480C. The critical issues like liquid metal corrosion of RAFM steel, tritium permeation into RAFM steel and magneto hydrodynamic drag generation due to flowing Pb−17Li is already reported. αAl2O3 + FeAl coatings have been found promising to mitigate these challenges & reported with a substrate of P91 steels. However, coatings with hot dipping process on RAFM is scarcely reported as RAFM steel is in under development. Hence, an experimental investigation is done to examine the effects of heat treatments to form FeAl with a top layer of Al2O3.
In-RAFM steel (9Cr-1.4W-0.06Ta) samples are hot dipped in a molten bath of 93%Al & 7%Si at 730C. These hot dipped samples are subjected to various heat treatments in 3 different routes. In the 1st route, normalizing heat treatment will carried out at 980C for 30minutes followed by thermal tempering in a muffle furnace at 760C for 90minutes. 2nd route consists of plasma tempering along with normalizing in which O2 will use at low pressure (5mbar) at 760C/90minutes with pulsed DC at -520V. Whereas in third route, to form α-Al2O3, plasma assisted heat treatments will get conducted directly without normalizing up to 24hrs. The transformations of these phases have been analyzed through X-ray diffraction. Moreover, to confirm Al2O3 coatings, its thickness as well as the case depth of diffused FeAl, cross section of coated samples were investigated by mapping and elemental depth profiling through SEM equipped with energy dispersive x-rays. Hence, this experimental study will address the phase transformations occurred during various heat treatments and its effect on formation of alumina films.

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P4.164 Technical background on design of DS recombiner for fusion facility

In a detritiation system, the function of recombiner is oxidation of tritium components. Detritiation system of a nuclear fusion facility should maintain its detritiation performance even in an event of accidents of the facility such as fire. The major technical background on recombiner design are detailed kinetics data, impact of gaseous impurities on kinetics, characteristics on pressure drop, impact of heat of reaction and effect of flow rate on kinetics. The volume of catalyst to be packed is evaluated by the overall reaction rate constant, required conversion efficiency and gas flow rate to be processed. The overall reaction rate constant has the effect of hydrogen concentration. Among the impurity gases that would be produced, water vapor, ammonia and halogenated acids affect tritium oxidation. It was demonstrated that the overall reaction rate constant is not affected by the scale of recombiner. Since the flow rate to be processed is large for the DS of a fusion facility, the amount of the catalyst to be packed inevitably increases. There is a strong need to set an allowable pressure drop so that the upstream line of the recombiner does not become pressurized. The height of the catalyst layer is determined from the allowable pressure drop. Then, the inner diameter of the recombiner is determined from the volume of catalyst and the layer height. In order to avoid tritium permeation from the recombiner surface, the adiabatic recombiner with a vacuum jacket can be applied. Consideration needs for the sudden rise of the catalyst temperature due to the reaction heat accompanying the combustion of the high concentration hydrocarbon gases. The maximum

temperature of the recombiner relies on the necessity of oxidation of tritiated methane. The rate of tritiated methane produced by side reactions on the catalyst between methane and tritium is negligibly small.

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P4.167 Mechanical design and first experimental results of a new multi-tube Pd-Ag membrane reactor for hydrogen isotopes separation in solid blanket concept of nuclear fusion machines

Pd-Ag membranes are well-proven technologies in nuclear fusion fuel cycle. Membrane technologies are one of the reference processes in the tritium extraction and recovery system of DEMO helium cooled pebble bed (HCPB) blanket. In the HCPB, a He + 0.1%w H2 purge gas is sent in the solid blanket in order to extract tritium. The result is a stream Q2 (Q = H, D, T) and Q2O species in a very low concentration and a huge amount of He purge gas. A pre-concentration stage allows to increase the Q2 and Q2O concentration while a following ultra-pure hydrogen isotopes separation will be performed via a Pd-Ag membrane module. For this reason, during the last few years, in the ENEA laboratories, a single-tube facility has been built and tested to assess the performances in hydrogen isotopes separation and water decontamination. In view of DEMO requirements, many efforts are now devoted to develop and optimise a scaled-up multi-tube membrane reactor aimed to perform both separation and reaction tests close by DEMO-relevant conditions. This work reports the mechanical design and optimization of a medium-scale (10-tubes) membrane reactor and the commissioning of a new facility built up in ENEA. In addition, the preliminary experimental results will be presented in this paper in order to investigate the behavior of the Pd-Ag membrane module by varying the main operating parameters (He/Q2 ratio, temperatures and pressures).

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P4.169 Neutronic analyses of the ITER TBM Port Plug with Dummy TBMs

To achieve the validation and testing of tritium breeding blanket concepts, mock-ups of breeding blankets, called Test-Blanket-Modules (TBMs), are tested in three equatorial ports of the ITER tokamak. Each TBM and its associated shield form a TBM-Set that is mechanically attached to a TBM Frame. A TBM Frame and two TBM-Sets form a TBM port plug (TBM-PP). Actually different TBM versions will be tested sequentially, each one tailored to the specific plasma operation scenario and test objective. In case a TBM-Set is not available, it can be replaced by a Dummy-TBM at any time. TBM Frame and Dummy-TBM are made of stainless steel and also need to meet ITER requirements on shielding, heat removal and safety. The maintenance operations for the TBM-PP include various activities in the Port Cell and in the Hot Cell. In order to assess the requirements, a detailed MCNP model of the TBM-PP, the Pipe-Forest (PF) inside the Port-Interspace (PI) and the Bioshield-Plug (BP) was developed and integrated into the ITER neutronics "C-Model".
This analysis concerns the neutronic behaviour of the TBM-PP with two Dummy-TBMs by means of activation and shut-down dose rate (SDDR) calculations. The analysis includes the evaluation of the radiation fields in the rooms around the TBM-PP inside the ITER tokamak at several cooling times. Two configurations scenarios have been considered, one with a PF structure inside the PI and one without this structure. The results show that the contribution to the SDDR from the TBM-PP at 12 days after reactor shut down is small compared to the contribution originating from the PF structure. Finally, the required design improvements are proposed in order to reduce neutron flux and SDDR.

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

and does not commit the IO as nuclear operator.

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P4.172 Fusion Technology Facilities & H3AT at UKAEA

The UK Government has invested ~€100M to create two new UKAEA centres for fusion research – Hydrogen-3 Advanced Technology (H3AT) and the Fusion Technology Facilities (FTF) both opening in 2020-21. FTF and H3AT will foster close cooperation with industry, academia and other international laboratories to develop and transfer knowledge between partners, offering opportunities to undertake R&D to reduce risk for ITER and to make significant contributions to the EU DEMO and international fusion programmes.

The FTF offers a complete development life cycle for materials and components in three facilities. The Materials Technology Laboratory develops and qualifies materials using small sample testing techniques to reduce costs and offer in-service testing. The Joining and Advanced Manufacturing Laboratory specialises in material joining and manufacturing technologies for fusion including additive manufacture and laser welding. It has a dedicated small sample test facility, HIVE, capable of providing up to 20MWm-2 over 20x20mm. The Module Test Facility provides fusion relevant testing environments, with heat flux up to 2MWm-2 (and higher localised flux) and DEMO relevant water cooling loop for metre-scale components.

At the heart of the H3AT facility will be a fusion relevant 100g tritium processing loop comprising of storage beds and distribution system, impurity processing system, isotope separation system and systems to detritiate water and air. These will service a flexible suite of enclosures and glove boxes to allow a broad range of tests and experiments. In addition the facility will contain a dedicated area to develop materials detritiation processes, a wet chemistry lab and a facility to handle other beta emitting gases.

This paper will describe the new facilities in terms of their technical capabilities and the progress to their realisation.

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P4.173 The CIEMAT LiPb Loop for Permeation Experiments

A new facility, CLIPPER, is being constructed at CIEMAT to investigate tritium extraction from PbLi. It consists in a forced circulation loop with the main objective of validating the technique of permeation against vacuum. Originally, CLIPPER was designed with two zones operating at different temperatures and connected through a recuperator, which gives most of the required thermal jump. In the cold leg the PbLi circulated at 300 ºC to have the PbLi in liquid phase, but high enough for avoiding cold points. The hot leg would increase PbLi temperature up to the experimental necessities (500 ºC). After a careful evaluation of the preliminary design, and considering the limitations of the laboratory in terms of power and available space, it was decided to change the approach to a simpler and cheaper facility. The new CLIPPER consists in a unique isothermal loop operating at 500 ºC. The main advantages are the lower PbLi inventory and the suppression of a very complex and expensive heat exchanger.
In CLIPPER, the PbLi is heated, melted and cleaned outside the loop, in a tank integrated in a dedicated glove box under argon atmosphere. The test section includes a prototype of permeator (TRITON) and its auxiliary systems, such as its associated vacuum system and instrumentation. TRITON consists of a rectangular multi-channel component with vanadium membranes. A novel gas injection system specifically designed for CLIPPER is used to introduce the gases that must be extracted through the permeator (hydrogen isotopes). This system, based on a multi-tube component with permeable niobium membranes, is able to solubilize the hydrogen in the PbLi up to the required

concentration.
The final design of CLIPPER is presented together with the integration of its main components, including a thermomechanical assessment of the loop.

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P4.175 Low pressure fusion exhaust gases separation

Plasma enhancement gases (PEGs) (such as: nitrogen, neon, argon and other inert gases) are injected into the plasma of several tokamaks in order to reduce the power load over the plasma facing component.
The exhaust gas in DEMO reactor consists of more than 90% of unburned fuel gas (D and T) and the remaining part will be He and impurities.
In DEMO reactor it is foreseen to recover the fuel gas and PEGs.
Experiments on separation of the exhaust gas from fusion reactors by using inorganic membranes have been performed.
The membranes with different geometry (single channel and multichannel) and pore sizes have been supplied by Ceramiques Tecniques et Industrielles (CTI SA) France.
Single gas permeation of air, helium (He), hydrogen (H2), nitrogen (N2) and argon (Ar) were measured at room temperature and feed pressure between few Pa and 100000 Pa.

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P4.176 Thermal analysis of a cryogenic distillation column for H2 isotopes separation

Cryogenic distillation (CD) of hydrogen in combination with Liquid Phase Catalytic Exchange (LPCE) or Combined Electrolytic Catalytic Exchange (CECE) process is used for tritium removal/recovery from tritiated water both for ITER/DEMO and for fission rectors like CANDU.
Tritiated water is being obtained after long time operation of CANDU reactors, or in case of ITER mainly by the Detritiation System (DS).
The ICSI cryogenic system consists of four distillation columns and significant effort is required in various batch mode operations for achieving high tritium concentration.
Some problems have been experienced with the forth column of the cascade regarding the heat transient transfer during start-up. This paper intent is to present a CAE thermal simulation of the column during the transitory thermal regime in order to investigate the temperature distribution along the column and its connections to other equipment. The simulation is done on the as-built design of the column and takes in consideration the actual configuration. This work provides an important platform to understand the thermal phenomena during cool down of a cryogenic distillation column besides finding the failures in thermal insulation or design flaws. The results of the simulation can be used as lessons-learned for future design work of cryogenic systems like Tritium Removal Facility from NPP Cernavoda or ITER/DEMO ISS.

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P4.183 Steam Generator mock-up design and testing suitable for Pb-Li technology demonstration and code assessment

One of the main challenges for DEMO is to overcome the Power Conversion System (PCS) issues for

a pulsed reactor. PCS is a complex system where the secondary circuits, connected to the Primary Heat Transport Systems (PHTS), should be integrated into an industrial and reliable system based, as much as possible, on proven technology. PCS should be designed to remove heat from PHTSs, during pulse phase and during the Dwell time, for a thermal power conversion to electricity, possibly with high efficiency.
Dual Coolant Lithium Lead (DCLL) concept is based on two coolant fluids flowing into two independent circuits: a helium circuit which oversees cooling the FW and some BB structural materials and a PbLi circuit removing the heat generated in the BZ. The heat from PbLi is used to produce steam by means of an innovative steam generator, that requires experimental tests to demonstrate the technological feasibility and performance evaluation.
The objective of the activity is to support the design of a LiPb steam generator and an experimental campaign defining the test matrix and the expected outcomes. Two configurations are analysed: the Steam Generator Bayonet Tube (SGBT) and the Helical Coil Steam Generator (HCSG). The first phase of the activity was dedicated to the evaluation of the steam generators performance, using a modified version of RELAP5/MOD3.3 (R5), which allows to identify the reference configuration for each typology. R5 has been modified to evaluate a correct two-phase heat transfer in helical coils.
After that, the CIRCE facility at ENEA Brasimone RC was selected to investigate the steam generators capability, carrying out the experimental campaign for the mock-ups of the two configurations. HERO (Heavy liquid mEtal – pRessurized water cOoled tube) was identified as mock-up for the SGBT and conceptual design of the HEX mock-up was carried out using R5.

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P4.177 Liquid-solid coupled interaction between thermal stress of FW and duct in CFETR blanket

The He-cooled Lithium Lead breeder blanket could be developed with the utilization of relatively mature material technology, which is used by Reduced Activation Ferritic / Martensitic (RAFM) steel as the structural material in China. It is necessary to analyse the first wall structure heated because of the thermal stress problems from two coolants in the blanket. The deviation from the thermal stress would directly affect the blanket life and the safe operation coefficient, and indirectly carries on affection of the enhancement of thermal efficiency from electricity generation.
The helium flow in the First Wall and LiPb flow with a transverse magnetic field in vertical channels in the blanket are investigated. The specially numerical MHD code based on the CFD software has been developed for analysis of the LiPb flow. The helium flow with four kinds of design scheme have been calculated and simulated. The three-dimensional temperature distributions of the LiPb flow in heating duct have been given. The analysis of the flow field and temperature gradient in the boundary layer of the duct have been performed. The heat transfer boundary condition of helium flow duct was determined by means of liquid-solid coupled method. The analysis for the structural stresses of the LiPb flow channel have been performed. The effect of the ratio of thermal load on the heat transfer characteristics of the helium and LiPb flow have been calculated and performed.

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P4.178 Development of a computational procedure to simulate activation of coolant/breeding loops in DEMO reactors

In future fusion DEMO reactors, coolant and breeding materials will flow along loops experiencing very different neutron irradiation environments, from those corresponding to in-vessel systems like blankets and divertor, up to regions without significant neutron radiation farther from the reactor. Moreover, in these loops there are material extractions due to the functioning of the tritium extraction and purification systems as well as release/deposition of activation corrosion products. In

these circumstances, the time evolution of the nuclide inventory in the fluids will depend on both irradiation history and material injection/extraction. A reliable prediction of this nuclide inventory evolution in the loops is a mandatory task for maintenance, safety and waste management analysis, as can be the most relevant radiation source in regions beyond the bioshield.
The aim of this work is to present a computational procedure to estimate the nuclide inventory evolution in this kind of loops. Such procedure is capable to simulate realistic irradiation conditions of the flowing materials, considering different spatial regions or blocks with the corresponding neutron spectra and residence times. In addition, the nuclide inventory can be modified at the entrance/exit of each block in order to simulate the abovementioned material injections and extractions.
The neutron fluxes used in the procedure are calculated by the MCNP5 transport code while the activation is carried out by the ACAB code. The coupling of these codes is managed by a script that joins the different blocks considered for the loop. In addition, because of the long computational time required for the simulation of the huge number of loops expected in DEMO, an extended effort has been done for computational optimization of the activation calculations based on the reduction of the rate equation matrix.

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P4.181 Computer based architecture to control Water Detritiation Process

Water Detritiation Systems (WDS) are used for CANDU (to remove tritium from heavy water) and also ITER. A water detritiaton facility was developed at ICSI Rm. Valcea based on Liquid Phase Catalytic Exchange (LPCE) combined with Cryogenic Distillation (CD). Initial application was for developing Cernavoda Tritium Removal Facility, but later these combined technologies were considered for fusion application (ITER WDS & ISS).
This paper presents control architecture of LPCE process from "Pilot Plant for Tritium and Deuterium Separation"(PESTD) ICSI Rm. Valcea, including hardware considerations and software implementation. To control the process it was used a concept based on DCS and SCADA application software that controls process and a complex architecture of servers, clients and operator workstations. The architecture of control system combined with hardware configuration and software applications avoid process control interruption and keep it within safety limits.
Dedicated faceplates to control equipment were developed to permit fine adjustments during transients and flexibility in change parameters of process. For safety purpose, in order to avoid human errors, specific authorization functions for operators were created together with interlocks to limit operator intervention.
This new system is constructed to respond to updated requirements for PESTD related to process configuration, safety (hydrogen safety in principal) and control of the process considering Romanian regulatory standards. The result of the work is complex as LPCE process is complex and can be used to any similar application. The expertise is used to complete all systems in PESTD using similar computer based technology. The approach used in PESTD can be considered for other similar facilities, the most significant change will be safety related as each facility has own safety requirements.

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P4.185 Study of tritium and helium generation and release from lead-lithium eutectics Li15.7Pb under neutron irradiation

At the moment, the lead-lithium eutectic was chosen as the material of the reactor blankets of ITER and DEMO. During the reactor operation radioactive tritium will be produced in the blanket, but the interaction parameters of hydrogen isotopes with lead-lithium eutectic are poorly studied. The most

significant processes affecting the release of tritium from the lead-lithium eutectic is its interaction with lithium atoms, as a result of which lithium tritide is formed. An increase in the temperature of sample above 300 ° C leads to a decrease in tritium release, which is caused by the appearance of local inhomogeneities in the eutectic, leading to a change in the ratio of lithium atoms to lead atoms in the sample. As a result, lithium atoms are deblocked with lead atoms (and, correspondingly, the increase in the chemical activity constant of eutectics), as a result of which the lithium atom can interact with the tritium atom. Physically, this process can be represented as an increase in the number of traps for tritium in eutectic with an increase in temperature. The main published works on eutectics were usually carried out with samples containing 17% of lithium, but even a small deviation in the lithium concentration relative to lead leads to significant changes in the properties of the eutectic with respect to hydrogen isotopes. That is why there is an urgent need for research on the generation and release of tritium from a lead-lithium eutectic with a lithium content of 15.7% directly under neutron irradiation.
In this paper, we present the results of experimental studies of the lead-lithium eutectic Li15.7Pb under neutron irradiation, as well as the parameters of tritium release from the lead-lithium eutectic Li15.7Pb for various temperature regimes.
The work was carried out within the framework of the MES of Kazakhstan project AP05131677/GF5.

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P4.186 Evolution of tritium transfers in the lithium loop of IFMIF-DONES

In the pathway towards the achievement of a demonstration nuclear fusion reactor DEMO, the construction of a neutron irradiation plant is a priority. Within the European framework, the construction of a facility that is capable of producing a significant amount of irradiation damage in materials as soon as possible was decided. The design for this Early Neutron Source (ENS) is DONES (DEMO-Oriented Neutron Source).

In DONES, a flow of liquid lithium is used as a target bombarded with deuterons in order to generate high energy neutrons for the irradiation of materials. The reaction between the deuterium beam and lithium produces, among other products, 3.9 grams of tritium per year.

Tritium as an unstable nucleus with beta decay must be controlled for safety. Target value of tritium concentration in the Li loop is below 1 wppm, while the target value for all H-isotopes is below 10 wppm.

A prediction of tritium inventories and leaks is fundamental to assure being below the safety limits. This is what has motivated the creation of models of these phenomena using the simulation tool EcosimPro. The program offers the possibility of mixing various disciplines (e.g., transport, control, hydraulics, etc.) by robust equation-solving algorithms. Starting from libraries previously developed, a DONES Li-Loop simulator of tritium transfers has been built.

The evolution of tritium inventories and permeation rates in the different components of the loop as well as a parametric study by H-trap efficiency are presented.

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P4.187 Tritium operations at the Laboratory for Laser Energetics

The Laboratory for Laser Energetics (LLE) at the University of Rochester supports a 35 kJ, direct-drive, laser to compress DT ice to study inertial confinement fusion physics. One millimeter diame-

ter, hollow, 6 to 10 micron-thick wall, plastic spheres are charged with deuterium-tritium (DT) gas mixtures up to several hundred atmospheres. These targets are cooled to cryogenic temperatures in a controlled manner to form single-crystal DT ice layers on the inner surface of the spheres. The DT targets are compressed under vacuum and cryogenic conditions by direct, uniform illumination with 60 beams of 351 nm laser light to achieve pressures in the several tens of gigabar range.

Filling the targets is a multistep process. Tritium is released from uranium storage bed and assayed to verify the working inventory. The gas is then compressed cryogenically and passed through a palladium purifier to remove decay helium-3 and tritium-beta generated impurities arising from tritium interaction with the stainless steel process lines. The purified gas is collected and compressed first cryogenically once again to approximately 100 bar and then mechanically to several hundreds of bar before the cooling process is initiated. The targets are filled by permeation through the thin wall spheres so the mechanical compression to the final pressure is slow to ensure the targets are not crushed in the process. Unused tritium is returned to the uranium storage beds.

The tritium inventory at LLE is limited to 1.5 g by New York State regulators. Emissions to the environment can not exceed 260 GBq annually. Several tritium capture technologies are deployed to intercept any tritium that has leaked or outgassed from the process systems. This presentation will outline the target filling process and discuss the technologies that are used to prevent tritium from being released to the environment.

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P4.188 Neutronics design analyses of helium cooled ceramic breeder blanket for CFETR phase 1

China Fusion Engineering Test Reactor (CFETR) is proposed to be the next generation fusion device of China. The conceptual design of CFETR has been completed and the engineering design is about to start. This work gives an overview of neutronics design analyses of helium cooled ceramic breeder blanket at institute of plasma physics Chinese academy of sciences (ASIPP), for CFETR phase 1 with the major radius of 5.7m and the minor radius of 1.6m. The effort involves 3 three different design schemes of helium cooled ceramic breeder blankets: the Breeding Unite scheme, the S-shaped cooling plate scheme and the U-shaped breeder tube scheme, the neutronic studies of which were all based on 3-D Monte Carlo simulation by MCNP and FENDL library. The tritium breeding ratios were shown to achieve the requirement, and the shielding performance was considered to be adequate to protect the superconducting toroidal field coil in inboard mid-plane, for all the 3 design schemes.

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P4.191 Development of PbLi facilities and experiments implement for fusion blanket technology

The liquid lead-lithium (PbLi) blanket concept has become a promising design for fusion DEMO and power plant reactors. To promote the successful application of fusion energy, some R&D activities on the PbLi blanket have been performed, such as structure material corrosion, thermal hydraulics, magnetic-hydrodynamic (MHD) effect, coolant impurities technology and LOCA/LOFA, etc.. Therefore, it is important to develop experimental facilities to perform the out of pile experiments and studies on these key issues before the engineering design of fusion reactor.
DRAGON PbLi experimental loops have been developed in China, including the thermal convection PbLi loops DRAGON-I (500°C), DRAGON-II (700°C), and the multi-functional liquid PbLi experimental loop DRAGON-IV (800°C, 2T). To perform the integrated experiments under multi-physical field conditions for DEMO blanket, the dual coolant thermal hydraulic integrated experimental loop

DRAGON-V was designed and finished the construction in 2017. The maximum flow rate of PbLi is 40 kg/s, the magnetic field is designed up to 5T. It is a unique test platform for the R&D of thermal hydraulic, material corrosion, purification technology and safety issues of liquid PbLi blanket to provide the necessary database for ITER-TBM and DEMO-TBM.
Up to now, some experiments have been conducted to investigate the key issues of PbLi technologies, such as corrosion behaviors of candidate structural materials with and without magnetic field, the PbLi alloy fabrication with high-level controlling of the impurities, purification technology of liquid PbLi coolant in the loop, MHD pressure drop test, and the interaction for typical coolants during accidents etc.. The results can support the development of the in-pile key techniques and components and the engineering design for ITER-TBM and DEMO-TBM, and also contribute to the final application of the advanced reactors.

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P4.200 Evaluation of high heat removal efficiency using cooling water channel with rib structure

A plasma-facing components (PFC) such as a first wall or a divertor are exposed to high heat flux due to high thermal radiation and high energy particles emitted from a core plasma. Various heat removal techniques had been proposed and developed to remove the high heat load of the PFC. We newly developed a high efficient cooling technique by inducing a turbulent flow in a cooling channel with the V-shaped staggered rib structure, which was originally developed to cool a target of an accelerator-based neutron source. The present work to pilot the future possibilities of the use of the structure of the water channel as a cooling system of the PFC.
The heat load experiments were performed by utilizing electron beam of 40kV in the Active Cooling Test stand 2 (ACT2) at National Institute for Fusion Science (NIFS). In the experiment, the cupper plate with nine water channels was set in a water jacket in the vacuum of the test stand and irradiated by an electron beam. The radiation thermometer and thermocouple set at the depth of 1.5 mm from surface was provided to measure the surface temperature of cupper plate. Temperature change of the cooling water was measured simultaneously. To evaluate the high heat removal performance of the “Rib channel”, it compared with the simple straight channel.
As a result of the heat removal experiments, we could confirm that the V-shaped staggered rib structure cam improves the heat removal performance more than double in comparison with the simple channel structure. In addition, the surface temperature increase of the surface was 100 K when heat load of 20 MW/m2 was supplied, so the “Rib channel” structure can be applicable for the water cooling system of the PFC.

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P4.203 Analytical and numerical calculations for the diffusion welding specimens on a Gleeble 3800 thermomechanical simulator

Diffusion bonding methods as a candidate solution for Plasma Facing Components in fusion reactors involved significant investigations over the last decades. For diffusion bonding methods the Gleeble 3800 thermomechanical simulator provides a different method for the diffusion welding process: instead of having a furnace with radiation heating and axial forces in a vacuum chamber – the heating is performed by Joule heating with 50 Hz alternating current passed through the specimens by grips at the ends. Gleeble System is a general-purpose servo-hydraulic thermomechanical instrument, it applies direct resistance heating up to 10,000℃/second and apply maximum static pressure 20 t [Uniduna, 2016] under mid 10-5 Torr range. However not in a high vacuum chamber, since the chamber has rubber O-ring.

The system gives opportunity to monitor along the specimen the heat distribution as the function of thermal conductivity, electrical resistivity and the thermal behavior of the joining surfaces during diffusion welding. The joining surface has varying thermal conductance during the welding process, in this way the diffusion welding could be observed during its welding process. The transient thermal distributions of welded specimens was also investigated using the different grips, jaw models of the Gleeble system with high or reduced thermal conductance.
This poster and paper intend to summarize the calculations of temperature distributions along the specimens in steady state and transient case during welding processes and the thermal aspects of the mating surfaces during diffusion bonding. Further goal of this work is to understand better the newly developed fusion material as ODS steels weldability in diffusion processes.

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P4.204 Development of ductile phase toughened tungsten for fusion plasma facing applications

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

We report studies on W-copper (Cu) and W-nickel (Ni)-iron (Fe) composites that were examined from both a composite development perspective and from a model material standpoint. Notched and un-notched bend specimens with various microstructures were tested at room and elevated temperatures in purified Ar at displacement rates ranging from 0.0002 to 2.0 mm/min. As expected, the general principles of DPT were observed in deformation and fracture of these materials. For example, the Cu phase effectively accumulated very large deformation and failed in a ductile manner in accordance with the behavior expected from its Ashby deformation map.

A finite element microstructural dual-phase model where the constituent phases were finely discretized based on digitized images and described by a continuum damage model was developed. In this approach homogenized meshed regions adjacent to a dual-phase meshed region of an actual bend specimen were created. The model is capable of modelling deformation, cracking, and crack bridging for DPT W-composites or any multi-phase composite structure where two or more phases undergo cooperative deformation. We simulated the stress-strain responses and fracture morphologies of W-Cu notched and un-notched specimens subjected to three-point bending and compared predictions to corresponding experimental results.

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P4.208 Mechanical alloying strategies for fabrication of new types of oxide dispersion strengthened low-activation one-phase high entropy alloys

The CoCrFeNiMn high entropy alloy (HEA) and its low-activation variants strengthened by dispersion of nano-oxides prepared via mechanical alloying were investigated. The two ways of oxide dispersion creation were verified: direct adding of oxides and internal oxidation of oxidizable elements by adding the gaseous oxygen to the alloyed powder. The grain refinement of the one phase FCC alloy by presence of oxides was determined by microstructural analyses to be 50%. The positive effect

of oxide dispersion on the strength properties was found to be between 30% and 70% for room and elevated temperatures, respectively. It was proven that the internal oxidation method can be used for a preparation of the ODS HEA alloy. This is especially applicable in the case of hard and abrasive oxides which do not erode during mechanical alloying and do not create fine dispersion.

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P4.209 The relationship between microstructure of ceramic coatings and γ-ray irradiation effect on deuterium permeation

Control of tritium permeation through structural materials is important in terms of fuel economy and radiological safety in fusion reactors; therefore, ceramic coatings have been researched as tritium permeation barrier for decades. Microstructural changes of the coatings, such as crystallization and deterioration, influence hydrogen isotope permeation behavior under reactor operation. In addition, recent research on deuterium permeation through metals under γ-ray irradiation indicated that deuterium permeation increased mainly by γ-heating. In this study, the γ-ray irradiation effect on deuterium permeation through ceramic coatings with different microstructures has been investigated.
Erbium oxide and yttrium oxide coatings were fabricated on reduced activation ferritic steel F82H substrates without heating by arc deposition and magnetron sputtering, respectively. A deuterium permeation apparatus was installed into a γ-ray irradiation chamber at Shizuoka University, and temporal changes of deuterium permeation flux under irradiation using a 26 TBq cobalt-60 γ-ray source were detected.
Deuterium permeation flux through the coatings increased during γ-ray irradiation at 250–300 °C, but noise was too large to detect changes in permeation flux at above 300 °C. The permeation flux of erbium oxide coating increased by crack formation during the permeation experiment at 500 °C. After deterioration of the coating, the permeation increase during γ-ray irradiation was approximately 0.5–0.6%, which was lower than that of the F82H substrate. On the other hand, the yttrium oxide coating had an amorphous structure and started crystallization at 500 °C. Although the irradiation effect before the crystallization was derived from the F82H substrate, the effect derived from the coating was detected after crystallization because the rate-determining process of permeation shifted to the coating. Compared the result at 300 °C before and after crystallization, the irradiation effect decreased to less than 0.5%, proving that the ceramic coatings reduced the deuterium permeation as well as the γ-ray irradiation effect.

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P4.215 Laser resistance of plasma-facing diagnostic components in ITER

The development of plasma-facing components with a high laser-induced damage threshold (LIDT) is an important part of R&D program for laser-aided diagnostics in ITER. A number of papers have been published studying LIDT of ITER materials. Most of them, however, are the investigations of integrity of first wall materials using laser radiation for simulation of pulsed plasma impact during transient events in ITER. Laser testing of in-vessel optical elements have been also reported, but were limited with a single laser impact without account taken of synergy effects during simultaneous/successive laser and plasma irradiation. The requirements to laser components of the Divertor Thomson scattering (DTS) diagnostics, driven by high-energy neutron and gamma radiation, intense particle fluxes and thermal loads, include also a high laser resistance throughout the diagnostic life cycle. The DTS laser elements are developed to withstand irradiation of the high-power probing Nd:YAG lasers (1064 nm and 946nm) with the pulse duration of 3 ns and energy of 2 J. The laser beam dump is the last element along one of the DTS probing chords absorbing laser irradiation to

prevent damage of the ITER construction elements. Fatigue degradation of tungsten, molybdenum and silicon, as possible materials of laser dump, were tested for the multi-pulse laser irradiation. The effect of plasma treatment and enhanced temperature on the LIDT has been assessed. The implementation of the laser dump for DTS and its performance under the high-number pulsed (up to 108 pulses) laser impact is discussed.

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P4.213 Ion irradiation in oxide nanoceramics: on the role of the irradiation spectrum at extreme damage levels

In the framework of future generation nuclear reactors, structural materials will face environmental conditions even more challenging with the highest radiation damage levels. To deal with this, oxide nanoceramics have been proposed. Oxide nanoceramics combine the enhanced radiation tolerance of nanocrystalline materials with the chemical inertness of oxides. In this work, the properties of Al2O3 films are evaluated as radiation damage approaches extreme values, reaching and even exceeding those anticipated for advanced nuclear systems (namely from 150 to 450 displacements per atom). Irradiation tests are performed using different ions in order to investigate the effect of the irradiation spectrum on the material's evolution. A comprehensive analysis of the irradiated samples is accomplished by X-Ray Diffractometry (XRD), Transmission Electron Microscopy (TEM), Scanning-TEM (STEM) and nanoindentation. The results show a general grain growth as the main structural change induced by irradiation in oxide nanoceramics. This structural change manifests mechanically through an initial increase of hardness (in accordance with the Hall-Petch relationship) and eventually through softening in the very last part of the experiment. Stiffness increases sublinearly with damage before reaching a plateau. Further, both hardness and stiffness depend on the phase present. A deep effort is made to establish a correlation between irradiation spectra and the evolution of the oxide's structural features and mechanical properties. The phase evolution appears to depend strongly on the ion utilized and on the irradiation spectrum, so the kinetic of the grain growth process. Finally, molecular dynamics simulations of displacement cascades are used to support the collected data and a preliminary model is formulated according to these observations. To conclude, an extensive characterization campaign is performed on nanoceramic Al2O3 in order to study its behavior at extreme radiation damage levels: structural and morphological changes are analyzed and supported by modelling.

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P4.219 A revised and expanded liquid metal property library for MELCOR

In order to facilitate the modeling of thermal hydraulic transients and accident scenarios in fusion reactors, a number of additional fluids were added alongside water in the original ATHENA code, which was based on RELAP5. The same libraries were later ported to both RELAP5-3D and MELCOR 1.8.6 for Fusion, both in use today. The libraries included a number of liquid metals of potential interest as tritium breeders, primary, or secondary coolants in fusion reactors, including Lithium, PbLi, Potassium, Sodium, and NaK. Thermodynamic properties of the liquid metals were generated using a five parameter "soft sphere" equation of state, for which parameter sets were available in the literature for the pure components.

This work describes a comprehensive update of the liquid metals property library, based on new

generalized Helmholtz potentials fit to experimental thermodynamic property data for each fluid, and a revision of the code that uses these to generate properties for MELCOR. The updates rectify certain inaccuracies in the existing libraries, both for the pure metals and the alloys, for which they were particularly acute owing to the approximate way in which the pure component equations of state were mixed. They also resolve some issues with code execution that occasionally arose as a result of the relatively coarse tables previously supplied to the code, and the manner in which empirical saturation relations were imposed on top of the equation of state. Finally, some new fluids have been added to facilitate the modeling of liquid metal plasma-facing surfaces, including liquid tin and SnLi alloys. The latter are used to illustrate some newly implemented methods to generate alloy properties from the pure component equations of state using suitable mixing rules. The new libraries will be available for MELCOR 1.8.6 for Fusion and (for the first time) standard MELCOR 2.2.

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P4.216 A Plan of Purification System for Lithium Loop of Advanced Fusion Neutron Source

A purification system for lithium loop is one of critical issues to design the accelerator-driven (IFMIF-type) fusion neutron source. It is written in IFMIF Intermediate Engineering Design Report that the levels of nonmetallic impurities such as nitrogen, oxygen, carbon, hydrogen isotopes and tritium should be less than 10, 10, 10, 10 and 1 wt.ppm, respectively, although there is little basis reported. To achieve the above impurities levels, especially nitrogen and hydrogen isotopes including tritium, it takes long time for purification before the main operation which generate neutron, and the data for these purification systems are so far insufficient to make an execution drawing. Therefore, a purification plan of lithium loop for Advanced Fusion Neutron Source (A-FNS) is suggested for early realization of A-FNS in Japan as below. In the suggested plan, the impurities levels of nitrogen and hydrogen isotopes including tritium are eased to 400 and 80 wt.ppm, respectively, while oxygen and carbon are the same as above. It is considered that the problems of nitrogen impurities would be plugging and erosion/corrosion, and those of hydrogen isotopes be plugging and tritium inventory in lithium. However, it is considered that the plugging would not be brought about, and the erosion/corrosion and the tritium inventory would not become critical to the lithium loop system under the operation temperature between 523 to 573 K and under the suggested impurities level. As the results, it is found that only a cold trap system for the purification system is necessary at first, and the hot trap system including nitrogen trap and hydrogen trap would become necessary after 4 or 5 years operation. That is, the adoption of the present suggestion could bring 4 to 5 years extension for the completion of the purification system in the execution drawing.

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P4.217 Fracture resistance study of tungsten wires as base material of fibre-reinforced plasma facing components

Together with the outstanding high-temperature mechanical and physical properties, the highest melting point of all the metals and thermal stability against recrystallization make tungsten (W) one of the main armour and heat sink materials candidate. Nevertheless, its applicability as a high-performance structural material is somewhat limited due to its typically brittle character at low temperatures and rather high ductile-to-brittle transition temperature. Among several ductilization strategies, one of the promising options is the development of W-fibre reinforced W-composite materials (Wf/W), where extrinsic toughening mechanisms can be used. For a detailed failure analysis of such a composite material, it is essential to obtain a complete understanding of fracture behaviour of drawn tungsten wires which are used as a base material incorporated in the tungsten matrix.

This contribution is oriented towards the study of damage tolerance of pure and potassium doped W wires with the main focus on the influence of different heat treatments. SEM with an EBSD detector was used to determine the microstructure of the wire. Vacuum annealing in the temperature range from 900℃ - 1600℃ permits the investigation of the microstructural stability of the two materials revealing a strong stabilization for K-doped wire with shifted recrystallization to above 1600℃. The RT fracture experiments were performed on both as-received and annealed wires, followed by a detailed characterization of the fracture surfaces. In contrast to K-W wire, the pure W experiences a drop in fracture toughness by about 70% after heat treatments, followed by a transition in failure mode from knife-edge ductile to the brittle cleavage fracture. The orientation sensitive experiments, conducted in two principle crack directions in respect to the drawing direction reveal significant anisotropy of fracture properties. Furthermore, the relationship between responsible deformation mechanism and degree of deformation is made by alternating the wire diameter between 50 and 150 µm.

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P4.218 Neutronics Analyses of the Bio-shield and Liner of the IFMIF-DONES Test Cell

IFMIF-DONES (International Fusion Material Irradiation Facility- DEMO Oriented NEutron Source) is an IFMIF-based neutron irradiation facility which aims at providing the irradiation data required for the construction of a DEMO fusion power plant. Comparing with IFMIF, DONES consists of only one deuteron accelerators (40 MeV and 125 mA), and utilizes only the High Flux Test Module (HFTM) for the irradiation of material specimens. The Test Cell (TC) in the IFMIF-DONES facility is the central confinement to envelop the end section of the accelerator, the HFTM, the lithium target assembly and some other lithium system components. A new design of the HFTM was recently developed for DONES, which has direct influence on the radiation distribution of the TC.

In this work, neutronics analyses have been performed on the bio-shield and liner, which are the essential confinements of the TC, with the updated HFTM design. The McDeLicious-17 code, which is an extension to the MCNP6-1.0 Monte Carlo code with the capability to simulate the deuterium-lithium neutron source in IFMIF-DONES, is employed in the calculations. Neutron and gamma flux, dose rate, nuclear heating, DPA and helium production have been calculated, using both a normal deuteron beam footprint size of 20 x 5 cm and a reduced beam footprint size of 10 x 5 cm. Using the normal beam size, the calculation results show that the high He production rate around beam level limits the possibilities of re-welding of the TC liner. The neutron dose rate in the Access Cell above the TC partially exceeds the limited for frequent access, hence additional local shielding is suggested. It is found that the nuclear distributions in the TC using reduced beam size are quite similar with that using normal beam size.

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P4.221 Experimental investigation of a Helium-cooled Breeding Blanket First Wall under LOFA conditions in view of code validation

The Helium Cooled Pebble Bed (HCPB) blanket concept is one of four EU DEMO (Demonstration Power Plant) blanket concepts currently under development. As part of the general strategy for the qualification of the design in view of licensing and operation, several experiments have been foreseen in to be carried out in the HELOKA facility at KIT.
This work presents the experimental results of a prototypical experimental set-up simulating a LOFA (Loss Of Flow Accident) in a helium-cooled Breeding-Blanket First Wall under typical heat-load conditions.

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P4.223 Optimization of knowledge transfer in ITER

Knowledge retention and transfer strategies are crucial for the success of every project. Especially on complex innovation projects, within high-tech sectors such as the aerospace industry or fusion, inter-organizational knowledge management needs to be promoted during the entire project life cycle. Project-created knowledge is based on the learning from the day-to-day activities, planning processes, reviews and evaluations coming from all involved stakeholders. In the case of IO, this includes in-house staff, Domestic Agencies (DA's), and contractors.
With cross-functional, inter-organizational and multicultural teams, ITER needs a robust framework to enhance the collaboration and transfer of expertise among the members and to promote the knowledge sharing. The capturing of lessons learned cannot be addressed as stand-alone process and must be integrated throughout the entire organization. As other Lessons Learned Information Systems (LLIS) have shown, an effective system aligns and promotes the dissemination of lessons learned, and embeds the process within the culture. Allocation of necessary resources for monitoring and follow-up is also crucial.
This paper shows how to optimize the lessons learned management on continuous and user-friendly basis beyond the functional and organizational boundaries. The methodology consists of a four-pillar framework, and some mechanisms to incentivize knowledge flow (and to prevent the organization from falling into poor documentation or not taking advantage of the learning effect). A description of the integrated concept and of the platform of knowledge transfer (with twofold goals of knowledge sharing at project and at sector levels) is provided in detail. Some constraints, individual and organizational barriers and risks are presented together with recommended mitigation actions.

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P4.224 Verification of SuperMC with HCLL mock-up experiment

Within the frame of European Fusion Technology Program, two neutronics mock-up (HCLL and HCPB) experiments have been performed using 14-MeV neutron generator (NG) to validate neutronics computational tools and nuclear data library. SuperMC, which is a nuclear design and safety evaluation software system developed by the FDS Team, was verified with HCLL mock-up experiment based on the latest neutron source, comparing with MCNP simulation in this work.
According to the content of HCLL mock-up experiment, the calculation of tritium production rate (TPR) and neutron/photon flux spectra at two positions were performed with SuperMC3.2 and MCNP5.1.60 based on data libraries JEFF-3.2 and IRDFF1.05. In order to ensure that the results obtained by the two codes are comparable, the parameter settings were kept the same in simulations.
The maximum deviations of TPR from 6Li and 7Li between the two codes were found to be ~0.65% ± 0.0022 (1σ) and 0.5% ± 0.0116 (1σ) at LiF TLDs. The discrepancies of total TPR (i.e. from 6Li+7Li) for natLi and 6Li with enrichment of 95% pellets ranged from -0.77% to 0.54%, and -1.17% to 1.04%, respectively. The deviations of integral neutron and photon flux are -0.03% and 0.47% at position A, and -0.02% and -0.04% at position B, respectively. In addition, a very good agreement for both neutron and photon flux spectra were also found.
The consistent results verified SuperMC's correctness in terms of fundamental physical processes, transport processes including biased simulation with cell/mesh-based weight window techniques, indicating that SuperMC can be applied in neutronics simulation of fusion reactors.

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P4.225 Development and Validation of Shutdown Dose Rate Calculation in SuperMC

Accurate calculation of the dose rate level around nuclear facilities after shutdown can provide important reference for the operation and maintenance of nuclear device, and also has important significance for the design of radiation shielding system and the disposal of nuclide waste. In this paper, based on the advanced neutron/photon transport calculation and activation calculation of SuperMC, an internal coupled rigorous 2-step (R2S) shutdown dose rate (SDR) calculation function was implemented.
The internal coupled method avoids tedious data transfer, which accelerated the whole calculation process. The capability to process complex models is enhanced through accuracy homogenization method for multi-material activation mesh based on CAD model. The materials for intersecting meshes which superimpose several cells are calculated by random sampling method using the bounding box data generated based on the CAD model before Monte Carlo particle transport calculation. The flexible sector cylinder activation mesh division method that supports direction angle sampling is implemented. It makes fine, accurate and convenient source specification automatically for any cylindrical mesh.
A validation assessment of internal coupled R2S SDR capability of SuperMC has been performed utilizing the FNG-ITER SDR experiment. The SDR calculation results of SuperMC were in good agreement with the measured values, the variation tendency of SDR along with cooling time calculated by SuperMC is consistent with experimental results. In summary, it preliminarily proved that the internal coupled SDR capability of SuperMC is correct, reliable and easy to use.

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P4.229 An optimized code system for activation and shut-down dose rate calculations

The neutron-induced activation of materials is an important issue for fusion facilities. Photons emitted by the activated material result in a photon field, whose spatial distribution must be taken into account when planning maintenance during shutdown and for decommissioning. One of the approaches to calculate the shut-down dose rate (SDDR) is the rigorous 2-step method (R2S) which is based on the data flow from neutron transport simulations to activation calculations to generate the photon source distribution. This source is used in subsequent photon transport simulations to calculate the dose rate distributions at different times post irradiation.

We present a new implementation of the R2S approach developed at KIT. Compared to the previous version, it provides a more flexible and clear user interface, assumes less simplifications and is written keeping in mind application to large problems thus establishing a more error-proof and performance-optimized workflow. In particular, the new implementation allows non-uniform rectangular superimposed meshes to represent neutron flux and material distributions. It provides a more robust method to extract information from MCNP model and utilizes FISPACT-II for the activation calculations. It allows computer-specific adjustments to FISPACT-II parallel invocation to optimize the use of file systems on cluster computers. To a large extent, standard Linux command line tools are used to handle intermediate data. Dynamic array allocation is applied in the modified MCNP for sampling the decay gamma source. The code with all interfaces are written in modern Fortran in a modular way, which allows subsequent improvements and modifications.

The paper describes in detail the new implementation and reports first results of the code-to-code benchmarking against validated R2S code implementations. Good agreement is obtained for the ITER SDDR computational benchmark. Further benchmarking against experiments is required to fully qualify the new implementation for applications in design analyses of fusion facilities.

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P4.232 Neutron flux uncertainty propagation in R2S-based shutdown dose rate calculations

Rigorous-Two-Steps (R2S) is one of the most important methodologies to estimate Shutdown Dose Rate (SDR) in relevant fusion facilities. This method is based on three calculations: first, a neutron transport calculation is performed to estimate the neutron flux in the facility during the irradiation phase; then, this neutron flux is used as input data for activation calculations, primary to enable the decay gamma source to be produced; finally, a gamma transport calculation is carried out in order to obtain the final SDR and radiological responses associated to decay gammas.
One of the critical issues of most R2S-based tools is that uncertainties of intermediate results (neutron flux estimations, isotopic inventory) can be significant but are not taken into account in the final SDR evaluation. In these conditions, it is very difficult to estimate the accuracy of the results obtained with the R2S methodology.
The principal aim of this work is the development and implementation of a methodology to enable the neutron flux uncertainty to be propagated to the final SDR, and consequently to allow the accuracy of R2S calculations to be estimated.
The propagation of the uncertainty is carried out explicitly in each calculation step using the standard error propagation law. The neutron flux uncertainty, assumed as known, is propagated to nuclear neutron induced reaction rates. Afterwards, the uncertainty of the decay gamma source is obtained, in a first approach, considering only direct parent-daughter reactions, although the gamma source is built taking into account all possible reactions. Finally, the gamma source uncertainty is combined with the gamma transport uncertainty to calculate the uncertainty of the final SDR estimated.
In this work, the methodology was implemented in R2S mesh-based tool R2SUNED, which couples MCNP5 and ACAB codes, and tested in ITER port plug shutdown dose rate benchmarking exercise.

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P4.234 Safety Important Classification of DEMO components

The identification of structures, systems and components (SSCs) performing safety functions is of a paramount importance for an EU DEMO development consistent with their implications in each phase of the project: design, fabrication, commissioning, operation, maintenance, inspections and tests. Moreover, this activity has to be performed at the early design stage for the correct definition of systems requirements, preliminary layout and integration. In addition, it may also be exploited as a supporting criterion for selection among the plant variants , e.g. alternative blanket concepts. Those SSCs assigned to a Safety Importance Class (SIC) will receive adequate attention during the different design, layout integration, construction, qualification and operational stages. The objective is to ensure and demonstrate that they will meet the minimum performance and reliability requirements throughout their intended lifecycle so that the safety function guarantee when required.

A graduation of safety important SSCs is adopted for DEMO on the basis of the IAEA Guide No. SSG-30, on SSCs classification. Three safety importance classes (SIC-1, SIC-2 and SIC-3) are identified for the SSCs on the basis of: a) the consequences the loss of their safety function could lead to, b) the involvement of the SSCs in preventing, detecting or mitigating an incident or an accident and, c) the involvement of the SSCs in bringing and maintaining the plant in a safe state. In this paper, the study performed for the safety classification of the Primary Heat Transfer System (PHTS) related to the Helium Cooled Pebble Bed (HCPB) and Water Cooled Lithium Lead (WCLL) blankets of DEMO is presented. From the results obtained it can be concluded that:
• HCPB PHTS SSCs can be classified as SIC-3 safety level for the confinement function;
• WCLL PHTS SSCs can be classified as SIC-2 safety level for the confinement function.

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P4.117 The influence of nitrogen on the deuterium retention within tungsten coatings for fusion applications

Plasma facing components (PFC) within a fusion device are subjected to a harsh operating environment such as high heat fluxes and exposure to high flux of hydrogen isotopes. This exposure can lead to a high fuel retention that can raise serious concern from safety point of view. One of the reason for the use of W as a material for construction of the first wall is aimed to reduce the fuel retention compared to carbon wall.
Nitrogen seeding, used in the operation of fusion equipment, represents a method to cool the divertor plasma and to reduce the W source in the divertor during inter ELMs. The effects of light impurity seeding have been analyzed so far mainly with focus on plasma heat transport and energy confinement and less on the plasma-wall interaction. However an exposure of the PFC to a combination of hydrogen isotopes and nitrogen can lead to changes in properties of exposed surfaces or to unexpected material behavior.
In the present work the influence of nitrogen on deuterium retention into the W coatings produced by high power impulse magnetron sputtering (HIPIMS) has been investigated.
NRA (Nuclear Reaction Analysis), XRD (X-Ray Diffraction) measurements, XPS (X-ray Photoelectron Spectroscopy) and SEM (Scanning Electron Microscopy) investigations were used for assessment of the chemical composition, fuel retention and the structure of the coatings. The elemental depth profile has been evaluated by using GDOES (Glow Discharge Optical Emission Spectrometry) measurement. A deuterium content up to 16 at.% has been obtained within the W coatings containing 30 at.% of nitrogen. It has been noticed that the N addition leads to an increase of the deuterium content within the coatings compared with W coatings obtained just in deuterium environment.

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P4.235 EU DEMO safety and balance of plant requirements. Issues and possible solutions

The DEMO preliminary safety and operating design requirements are being defined aiming at obtaining the license with a relatively large operational domain.
The DEMO design approach is being organized, by taking into account the Nuclear Power Plant ITER and Generation IV lesson learnt. Outstanding challenges remain in areas exhibiting large gaps beyond ITER. Those require a pragmatic approach, especially to evaluate and improve the readiness of technical solutions through dedicated physics and technology R&D.
Therefore, a system engineering approach is adopted based on an integral plant design analysis which follows immutable goals as safety, availability and power provision to the grid. The immutability of the latter ensures not only the identification of critical interfaces but also the margin of possible solutions and, moreover, the definition of target parameters for technical systems to be met in order to arrive at a feasible DEMO.
The overall DEMO plant design has to be strongly safety and operation-balance of plant oriented; which represents a significant change of the culture in the fusion community.
The paper describes a few important aspects of safety and balance of plant that require early attention and a continuous reanalysis at any significant design change. This includes: (i) safety provisions required by the coolant options following some reference accidents; (ii) tritium inventory limit control considering the substantial throughput; (iii) permeation of tritium through the Primary Heat Transfer System; (iv) conditions for a plasma shutdown, (v) pulsed operation and relevant interfaces with the grid and BoP systems; (vi) layout of the tokamak building to accommodate Remote Maintenance meeting layout and safety criteria.
Any effort to reduce the complexity of a Fusion Power Reactor design through simplification and rationalization of the design and operation of the main systems translates into a more robust plant configuration enlarging safety margins and operational thresholds.

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P4.236 Cernavoda Tritium Removal Facility – tritium supplier for fusion facilities

ICSI has completed in 2015 the conceptual design of the Cernavoda Tritium Removal Facility (CTRF). CTRF is located at CNE Cernavoda, a NPP subsidiary of SNN Bucharest, and is sized to process heavy water from 2 CANDU reactors, treating 40 kg/h heavy water over 40 years with a detritiation factor of 100. CTRF removes tritium using liquid phase catalytic exchange (LPCE) paired with cryogenic distillation (CD). The design of CTRF uses expertise from ICSI (Pilot Plant for Tritium and Deuterium Separation) and from Canada (Kinectrics), together with experience of experts from Wolsung TRF project, research tritium laboratories and industry. CTRF project is the most updated project for a TRF, including last safety requirements for a tritium industrial facility.
The objectives of CTRF will be to reduce operators' doses and tritium releases during refurbishment of units and normal operation, later to support decommissioning of the reactors by reducing the tritiated heavy water quantities which else will be radioactive waste.
Construction of CTRF starts in 2018 and it is scheduled to begin detritiation of heavy water from reactor U1 in 2025. After this date tritium stored will become available to be used for fusion research and industrial facilities.
First phase of detritiation process is to reduce tritium content from 65Ci/kg to 10Ci/kg, second phase to maintain the tritium concentration up to 10Ci/kg and third phase to use CTRF to reduce tritium content as low as possible before decommissioning of the site.
The paper will present the technologies used, a prediction of tritium production and some considerations about possible use of CTRF for He-3 supply, as by-product of detritiation process.

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P4.032 Electromagnetic analyses of Single and Double Null configurations in DEMO device

The development of a conceptual design for a demonstration fusion power plant (DEMO) is a key priority of the recent European fusion program. The DEMO design faces an even higher challenge taking into account that, compared to ITER, the European DEMO design has a fusion power that is four times higher and a major radius that is only 1.5 times larger than ITER. From a first review of the wall loads and the associated limits in DEMO, the power load on the Plasma Facing Components (PFCs) associated to the single null configuration clearly underlines a significant challenge that requires substantial engineering efforts as well as the possibility to explore alternative divertor concepts.
In this paper the exploration of double null configurations in DEMO is proposed. Indeed, a comparative electromagnetic analysis of single and double null configurations is performed in terms of vertical stability (VS) and disruptions. VS properties and disruptive events are highly relevant for the protection of the PFCs and they also determine the maximum tolerable elongation of the plasma, to which the performance of the device has an extreme sensitivity for fixed major radius.
The comparison of SN and DN configurations is not simple and it also requires that the machine design is optimized to a comparable level. As the DEMO SN baseline design is largely based on ITER and hence already optimized in terms of vertical stability, special attention has been given to reducing the distance between toroidal conducting structures and plasma in DN. The assessment of the VS performance is done in the 2D axisymmetric case in terms of best achievable performance assuming a constant voltage on the vertical stabilization system. The disruptive events are analyzed in the 3D case to take into account the non-axisymmetric effects related to port locations and non-toroidally continuous structures.

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P4.119 Enhancements in the structural integrity assessment of plasma facing components

The preferred design of the ITER divertor target component is the monoblock, comprising a tungsten armour block through which passes a CuCrZr cooling pipe joined to the tungsten via a copper interlayer. This construction currently looks to be one of the favoured designs for DEMO. Ideally this structure can be assessed for structural integrity by finite element (FE) analysis using the ITER structural design code (SDC-IC). However, the multiple materials used in monoblock construction introduces many factors causing difficulties in the application of SDC-IC. The factors include residual stress, dissimilar material joints, the interaction of pipe and interlayer plasticity, the effects of irradiation hardening and the possibility of tungsten recrystallization. This presentation discusses some of the FE simulation methodologies that can be employed to overcome these issues. The effects of residual stress (caused be the differential contraction during manufacture) can be included in the stress assessment by using elasto-plastic methods and a simulation step representing the manufacturing cycle. The effects of dissimilar joints (which create singularities) can be avoided by modified joint design or hot spot methods. Irradiation hardening effects can be simulated directly by applying modified material stress/strain characteristics within an elasto-plastic simulation. The ratchetting effects of pipe/interlayer plasticity interaction can be addressed by separating material ratchetting from structural ratcheting, and the effects of tungsten recrystallization can be included by modification of modelled material characteristics and the assessment of plastic strain above and below the materials DBTT. Using these methodologies, analysis results can be assessed against the current list of SDC-IC failure mechanisms, allowing an improved assessment of structural integrity to be made. These methodologies form part of a divertor component design plastic analysis procedure being developed for EUROfusion. It is intended that they can be applied in the analysis of all types of multi-material plasma facing components.

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P4.132 Production of tungstenfibre reinforced tungsten composite by a novel continuous chemical vapour deposition process

For the use in a fusion reactor, tungsten has unique properties such as a high melting point, low sputter yield and hydrogen retention as well as moderate activation. The brittleness below the ductile-to-brittle transition temperature and the embrittlement during operation are the main drawbacks for the use of pure tungsten in plasma facing components. Tungsten fibre-reinforced tungsten composites overcome this problem by utilizing extrinsic mechanisms to improve the toughness. Dense samples (>99%) with the size of 50x50x3 mm^3 incorporating nine layers of tungsten fibre fabrics have been successfully produced by layer wise chemical deposition, where tungsten is deposited on hot surfaces from gaseous precursors on fibre preforms. The sequential deposition process with intermediate vents for the fibre placement could lead to internal interfaces in the tungsten matrix due to surface oxidation. Despite this caveat, small specimens for mechanical testing could be fabricated from the samples, which revealed the expected improved fracture toughness. The next step in the material development is the production of larger components and testing under cyclic high heat flux loading. A dense matrix produced with a reliable production routine is one of the major issues for the production of such components. A novel continuous deposition process was introduced by designing and testing a set up For Rotary Enhanced Deposition (FRED) in the already existing W InfiLtration MAchine (WILMA), fabricating first samples. These samples were examined by microstructural analysis and compared to material produced by the standard technique. It is shown that this advanced set up in combination with tungsten fabrics allows the productions of tungsten-fibre reinforced tungsten composite in a considerably faster and more defined way.

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P4.007 Neutral beam assisted plasma breakdown on tokamak COMPASS

Using helium as a working gas in COMPASS tokamak is a vital part of the research program. Unfortunately plasma breakdown in helium is quite difficult owing to heliums high ionization potential. In the absence of impurities with lower ionization energies helium plasma breakdown is very sensitive to start-up parameters and thus unreliable. To mitigate this issue several methods have been proposed. Using Neutral Beam Injectors as a source of ionized particles is an elegant solution. COMPASS is equipped with two neutral beam injectors. Both can deliver up to 350 kW of neutral particles with energies of up to 40 keV.
In this contribution we present the results obtained during Neutral beam injection assisted plasma breakdown in deuterium and helium plasmas. Breakdowns have been successful at lower loop voltages than without neutral beam injection. Timing of the breakdown was also more consistent. Since installed neutral beam injector use cryo condensation pumps, the argon frosting technique has been used to facilitate sufficient pumping of Helium for reliable operation of the injectors. Performance of the argon frosted cryopumps is also reviewed

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P4.008 Support structure of the COMPASS-U tokamak: validation by ANSYS modelling

A new high magnetic field tokamak, COMPASS-U [Panek et al., Fusion Eng. Des. 123 (2017) 11 – 16], replacing the currently operating COMPASS tokamak at IPP Prague is being designed and constructed nowadays. As a result of high magnetic field in the machine (B0 = 5 T) large forces (up to 6.5 MN) acting especially on the TF coils are expected. In order to keep the loading of the coils, which will be made of OFHC copper, low enough to prevent material failure and to ensure their high cycle-life, massive external support structure was designed. The outer support structure will be made of the stainless steel plates of 200 mm width.

The numerical validation of the support structure design was performed by means of ANSYS software calculations. Electromagnetic forces for several worst-case scenarios acting on the TF and PF coils and on the coils of the central solenoid were evaluated using ANSYS Maxwell code. The resultant volume force density was consequently used as a load in mechanical stress distribution calculation by ANSYS Mechanical.

The results revealed that the suggested support structure is fully able to capture vertical separating force acting on the TF coils. As a result of the interaction between the TF coil currents and the poloidal fields significant torsional shear stress in the TF coils was evaluated. High pressures between the coils of the central solenoid and TF coils were also observed. Critical places were identified and structural design changes were suggested to keep the mechanical stress below the acceptable limits.

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P4.056 Integration of the TESPEL injection system at W7-X

Since impurities determine the plasma performance in magnetically-confined fusion plasmas, analyzing their behavior and controlling their confinement is significantly important for a stable and steady-state operation of fusion devices. A comparison of the impurity transport in the core region

of large stellarators, such as Wendelstein 7-X and Large Helical Device (LHD) is quite useful to gain a better understanding of the impurity transport in reactor-relevant stellarator plasmas. Therefore, a Tracer-Encapsulated Solid Pellet (TESPEL) injection system has been newly installed for the operational phase OP1.2b of W7-X. The injector was designed and manufactured by NIFS to achieve a similar TESPEL deposition location as in LHD. Technical guidelines and functional specifications, addressing the specific conditions at W7-X had been implemented in close collaboration between NIFS and IPP. This resulted in a compact 3-stage differential pumping system considering spatial restrictions due to existing and future diagnostics, material requirements and limitations due to the magnetic stray field in the vicinity of the turbo pumps.
This contribution reports on the design and the integration of the new TESPEL injection system to W7-X and first achievements during the commissioning of the system.

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P4.059 Refurbishment of JET magnetic diagnostics

In a tokamak device magnetic diagnostics play a key role in the understanding of plasma physics, for control and safe operation. JET tokamak has hundreds of magnetic sensors distributed over the torus, designed to withstand neutron fluence. By the end of 2016 experimental campaign C36B, JET lost several pick-up coils used both for equilibrium reconstruction (slow coils) and MHD analysis (fast Mirnov coils). Slow coils are based on a MIC (Mineral Insulated Cable) wound around a stainless steel former, while the Mirnov coils are based on non-insulated cable wound around a ceramic alumina former. In order to restore the JET MHD modes analysis capability for the coming DT campaign, 27 faulty in-vessel coils were refurbished. A new design was proposed to try to diminish the failure rate, mitigating the possible cause that led to those coil failures. New sensors will use Glidcop® wire wound around a ceramic former, to withstand chemical reaction coming from gas species used in the experiments and to better transfer heat load. The wire diameter was increased and a ceramic coating inside the coil casing was introduced to improve electrical insulation. To reduce cost of future intervention all new coils are remotely-handleable (RH) and no in-vessel manned intervention is required. Prior to installation a calibration procedure and electrical tests were carried out in the laboratory. The calibration was performed using a Helmoltz coil together with a known reference coil to determine NA at 0 Hz (DC) and the transfer function amplitude, with respect to the reference coil, up to 200KHz. Two of the 27 newly manufactured coils could not be installed due problems on the existing in-vessel cabling. The commission of new coils will use dedicated pulses and a FEM 2D axial-symmetric code to predict the waveform and compare it with the experimental ones.

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P4.019 Development of High-Power Pulse Neutral Beam Injection System for the Versatile Experiment Spherical Torus

A short-pulse and high power neutral beam injection (NBI) system is developed for the Versatile Experiment Spherical Torus (VEST) as a main plasma heating device. The NBI system is designed to inject above 0.5MW neutral beam heating power at the hydrogen ion beam energy of 20 keV in the pulse length of 10 ms. In addition, a design feature of VEST NBI system is changing the beam injection energy in the pulse duration by utilizing a pulse power system based on 2-stage Marx generator for the ion beam extraction and arc plasma generation. Change of the incident ion beam energy is required to reduce neutral beam losses resulting from the shine through and orbit loss under the low initial target plasma density and toroidal magnetic field strength of the VEST. Based on the design parameters, high power arc ion source and power supply system for VEST NBI system are successfully developed and commissioned.
In this paper, a summary of VEST NBI design features is described, and experimental results of pulse

arc ion source and neutral beam transport in various operating conditions are presented.

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P4.093 An arc electrical simulation method for superconducting magnet system in safety-related fault condition

In large superconducting magnet system, the release of inductive energy stored inside the superconducting coils provides a considerable potential of dc-arc hazards in case of an accident such as unmitigated quench. Safety analysis and numerical calculation have raised great concerns. This paper will present an electrical arc simulation method using Matlab/Simulink software to simulate the electrical responses with arcs presence in a 'global' circuit network. The inline arc and outline arc simulations in ITER TF electrical circuit integrated with two arc models will be adopt as examples to simulate the transient electrical behavior of TF coils and arc power during some faulty conditions. We developed the model that considered the arc length variation and arc ignition. Kronhardt model and High Pressure Arc model are compared through the simulation results. The proposed simulation method shows the convenience and efficiency of the arc electrical simulation. It can be used for further thermal damage assessment and arc detection research.

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P4.021 Commissioning of vacuum pumping system for the second KSTAR neutral beam injection system

To achieve the high performance plasma in the Korea Superconducing Tokamak Advanced Research (KSTAR) tokamak, Neutral Beam Injection (NBI) system has been installed. The first NBI (NBI-1) was installed in 2010, which provides a 100 keV deuterium neutral beam of 5.5 MW maximum using three ion sources. The second NBI (NBI-2) with another 6.0 MW will be constructed until 2019. In this process, one ion source will be used to improve the plasma performance at 2018 KSTAR campaign. As the vacuum condition can have substantial effect on the beam performance, the vacuum pumping system for NBI-2 has been carefully designed. The cryopumps for NBI-2 were designed and the overall vacuum pumping system of NBI-2 was installed including turbo pumps, mechanical booster pumps, and dry pumps. In this paper, the detailed engineering design, fabrication and installation results of large condensation cryopump and vacuum sub-components were explained. The effective pumping speed and pressure distribution will be evaluated through the commissioning of vacuum pumping system for NBI-2. Finally, we plan to establish the operation procedures of the vacuum pumping system.

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P4.099 Optimization of the cooling capacity of the cryo-magnetic system for EU DEMO at the pre-conceptual design phase

The thermal hydraulic analysis of the DEMO cryo-magnetic system has the main objective to minimize the refrigeration power. The cryo-magnetic system includes the superconducting magnets cooled by forced flow supercritical helium at about 4.4 K, the cryo-distribution lines and valve boxes, and the cryogenic system with several cold boxes. The DEMO cryogenic system is at the pre-conceptional phase design, with the identification of the heat loads requirements of the main

cryogenic users. The analysis will focus on the cooling capacity of the superconducting magnets, which are the main contributors of the 4.4 K heat loads. Simcryogenics, a dynamic modelling tool developed by CEA, has been used to model a supercritical helium loop for cooling superconducting magnets, driven by a cold circulator and exchanging the heat loads with a liquid helium phase separator (cold source) pumped down with a cold compressor. An optimization algorithm has been developed to optimize the cold source temperature and the supercritical helium mass flow, in order to minimize the refrigeration power. The refrigeration power covers the cooling capacity to absorb the heat loads of the superconducting magnets (nuclear heating, thermal radiation and conduction) but also the pumping powers from the cold circulator and the cold compressor. The optimization problem is subject to the requirement of keeping a temperature margin of 1.5 K with respect to the critical temperature of the superconducting material. The optimization has been applied on one cable in conduit designed by CEA, but could be relevant for comparing different designs and their cooling requirements. At this pre-conceptional design phase, the interest of such analysis is to contribute to the optimization of both the cryogenic system and the superconducting magnets to minimize the refrigeration power, one of the important investment and operation costs for the future tokamak.

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P4.137 Design and analysis of dual arm robot for the intelligence maintenance of CFETR

The China Fusion Engineering Test Reactor (CFETR) is the testing fusion device which would be the prototype for future commercial reactor. However, the traditional maintenance way is mainly remote handling which is time-wasting and low efficiency. To meet the demands for more complicated maintenance, it is no hesitation to start the more intelligent devices for the fusion device. The dual arm robot is a part of the intelligent devices which is mainly responsible for some delicate mission, for example, bolting, welding and snatch. These actions cannot be completed by the single robot because of its operation complexity. Consulting with the previous experience, the RH team in Institute of Plasma Physics Chinese Academy of Sciences (ASIPP) and Lappeenranta University of Technology (LUT) has studied the actual usage and designed the dual arm system together for the CFETR. This paper will introduce a dual arm robot system which will be shown about its design and analysis range from the mechanism to the system frame work, from control strategy to the task-based algorithm briefly.

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P4.109 Effect of Initial Condition on Seismic Analysis of the HCCR TBM-Set

In this paper, effects of initial condition on seismic analysis for the HCCR TBM-set are evaluated using finite element analysis. Because of difficulty to predict when an earthquake occurs during operation, various scenarios are considered in the structural integrity assessment in ITER. To perform the simplified analysis, it is important to understand the effects of initial conditions on the structural analysis for seismic loads. In case of the seismic analysis for the HCCR TBM-set, response spectrum analysis based on modal analysis are performed. In this paper, effects of initial condition on modal analysis and response spectrum analysis with a simple pipe model are investigated and a seismic analysis approach for the HCCR TBM-set is proposed considering initial condition effects.

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P4.025 Approaching final design of the ITER EC H&CD Upper Launcher

The ITER ECRH system consists of 24 gyrotrons and their associated power supplies providing up to 24 MW millimeter wave heating power at a frequency of 170GHz, a set of transmission lines connecting the gyrotrons with the equatorial and the four upper launchers. With its high frequency this heating system provides the unique capability of driving locally current due to the small beam focus of just a few centimeters. While the equatorial launcher mainly acts for central current drive and current profile shaping, the upper launchers aim on suppressing MHD instabilities, especially neoclassical tearing modes (NTM) to avoid plasma disruptions triggered by NTMs.

The Upper Launchers inject the millimeter waves through a quasi-optical section consisting of three fixed mirror sets and the front steering mirror set. The eight overlapping beams have focal points optimized for suppression of the q=3/2 and q=2/1 NTMs.

Due to several project change requests the Upper Launchers and the connected ex-vessel system were redesigned. The changes include a new boundary geometry of the launchers as well as a newly designed cooling system for the Blanket Shield Module (BSM), a modified flange of the BSM to the structural main frame and a refined optical design. Additionally shield blocks with integrated in-vessel waveguides were redesigned as well as the closure plate with waveguide and supply line feedthroughs. The changes further include newly designed ex-vessel waveguide components with a reduced aperture and redesigned ultra low-loss CVD diamond windows. Finally several components originally foreseen as off-the-shelf components have become part of the design scope.

The new launcher design status is presented with selected results on numerical design validation.

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P4.001 Coupled seismic analyses of the ITER Tokamak Complex building and Tokamak machine

ITER is probably the most ambitious energy project in the world today, whose main objective is demonstrating the scientific and technical feasibility of nuclear fusion as an energy source. 35 nations are collaborating to build the world's largest tokamak fusion reactor, a magnetic fusion device that has been designed to prove the feasibility of fusion as a large-scale and carbon-free source of energy.
The ITER design must face a large variety of potential threats and complex loads, including seismic events. These are a design driver for many of the ITER buildings and main mechanical components, including the Tokamak machine and the building that supports it: the Tokamak Complex. The Tokamak Complex is a reinforced concrete building that houses the 23000-ton Tokamak machine, where the fusion reaction will take place. The Tokamak Complex is more than 70 m high, its plan dimensions correspond to those of a standard football stadium and it will become one of the largest seismically isolated structures ever built.
As a nuclear facility, the seismic design of ITER must ensure the corresponding seismic safety requirements are met according to the French regulation. On the other hand, too conservative approaches may have important technical and financial consequences.
This article provides a global summary of the latest works carried out by the European Domestic Agency, Fusion for Energy (F4E), for the definition of the seismic environment within the ITER Tokamak Complex, including the derivation of seismic floor response spectra as well as the characterisation of the complex interface seismic forces between the Tokamak machine and the building, which are critical for a correct design of the corresponding supporting elements.
This paper does not commit IO as nuclear operator.

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P4.003 The study on blowout system of EAST upper divertor

The divertor which is used to discharge reaction energy is the core component of the EAST Tokamak. The divertor is cooled by cooling water when the EAST machine is in operation. In order to prevent the cooling water from corroding the components, and to avoid the uneven baking temperature caused by the cooling water residue during the next baking, the cooling water inside the divertor pipe needs to be drained after operation. Due to the special installation and structural features of the upper divertor, cooling water is gravity drained with residual water still trapped in the system, the effective method is that the upper divertor should be blown out by pressurised nitrogen gas. The performance of blowout system is determined by the pressure difference between the inlet and outlet of pipeline. The operating dates show that under the current operating pressure of the system, there is still too much cooling water in the cooling pipe, which needs to be further studied.
In this paper, the FLUENT fluid simulation calculation method is used to analyze the influence of the differential pressure at the inlet and outlet of two in series divertor modules. To study the relationship between the amount of water in the pipeline and the pressure difference, the pressure difference at the inlet and outlet is fixed, and blowout system is performed through multiple sets of pressure differences. The simulation and experiment results prove that the cooling water in the upper divertor is less retained in the cooling pipe when the pressure difference at the inlet and outlet is 0.8 MPa, which meet requirements of the EAST completely.

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P4.005 TRANSP-based optimization for tokamak scenario development

An optimization approach that incorporates the predictive transport code TRANSP is proposed for tokamak scenario development. Optimization methods are often employed to develop open-loop control strategies to aid access to high performance tokamak scenarios [Fusion Eng. Des. 123 (2017) 513–517]. In general, the optimization approaches use control-oriented models, i.e. models that are significantly reduced in complexity and prediction accuracy as compared to physics-oriented prediction codes such as TRANSP but have the advantage of enabling fast simulations. In the presented approach, control-oriented optimization is combined with TRANSP-based optimization. The fast control-oriented optimization accelerates the TRANSP-based optimization by providing an initial guess solution, and the TRANSP-based optimization improves the accuracy of the control-oriented optimization by providing updated model parameters on each iteration. As a test case, the injected neutral-beam power, high-frequency fast-wave power, and electron-cyclotron heating power are optimized to develop a control strategy that maximizes the non-inductive current fraction during the ramp-up phase for NSTX-U. Simulation studies towards the achievement of non-inductive ramp-up in NSTX-U have already been carried out with the TRANSP code [Nucl. Fusion 55 (2015) 123011 (12pp)]. The optimization-based approach proposed in this work is used to maximize the non-inductive current fraction during ramp-up in NSTX-U, demonstrating that the scenario development task can be automated. The proposed optimization technique has the potential of playing a critical role in achieving a robustly stable non-inductive ramp-up, which will ultimately be necessary to demonstrate applicability of the spherical torus concept to larger devices without sufficient room for a central coil.

This work has been supported by the US Department of Energy under DE-SC0010661 and the DOE Office of Science Graduate Student Research (SCGSR) program award (http://science.energy.gov/wdts/scgsr/).

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P4.010 Increasing the integrated power input in W7-X

In 2017 the Wendelstein 7-X stellarator (W7-X) has performed the second experimental phase (OP 1.2a), introducing divertor operation with a inertially cooled test divertor unit (TDU), and graphite-covered first wall.
The divertor operation resulted in a substantial improvement of the plasma parameters and the inertially cooled TDU and the graphite walls also allowed much longer discharges with higher heating power. This could be observed by the possibility to increase the energy input up to 80 MJ, compared to the 4 MJ in the first phase (OP 1.1) without TDU. This was a first step in the mission of W7-X to achieve steady-state discharges at a high heating power.
This heating energy increase has been performed in several steps, in a dedicated procedure, identifying hotspots of localized energy deposition with thermography cameras and measuring the instantaneous temperature increase on the target plates, baffles and the walls as well as the slow temperature increase of divertor structures. Further, the plasma impurities have been monitored during the increase of the integrated power deposition in the device to detect a possible overloading of the first wall. The increase of integrated power deposition in the plasma was strongly depending on the magnetic configuration. The trim and control coils turned out to be good tools to adjust the wall loading. Procedures to conduct this increase in the integrated power deposition and the results for several magnetic configurations will be discussed.

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P4.014 The 3.7GHz LHCD system on HL-2M tokamak

A 3.7GHz lower hybrid current drive (LHCD) system was built on HL-2M in 2017. A Full-Active Multijunction (PAM) launcher is installed. The peak parallel refractive index is 2.25 with a range from 2.1 to 2.4. ALOHA code predicts a low Reflection Coefficient (RC) within a large range of the HL-2M plasma density. TE10 to TE30 mode converters are designed for the antenna to divide the power in three parts equally in ploidal direction,while the power is divided to 8 equal parts in toroidal direction. As a result, There are 192 active grills (6×32) facing to the plasma, with passive ones at each side.The antenna is fed by four high power pulsed klystrons TH2103C1. The klystrons are protected from the reflected power by high power circulators. The RF power is transmitted to the launcher in TE10 propagation mode through the rectangular waveguide (WR284) transmission lines. The transmission line is pressurized with 2 bar of nitrogen to prevent arcing. The 4 klystrons are fed by a pulse step modulation (PSM) high voltage power supply (HVPS). The fast switch-off time is less than 10 microseconds.
With the help of the power recombiner, the power capacity of each transmission lines reaches 500kW. The recombiner is based on the microwave power coherent synthesis principle. It consists of two rectangular waveguides with the shared height wall removed. Simulation and optimization is performed by finite element method. The measured results agree with the simulated ones within the error range. The power combining efficiency reaches 98.1% and is extended when the input power is not balanced.
The transmitter was commissioned. The total output power exceeds 2.02MW with a 3s duration after the main parameters were optimized. The output characteristic of the transmitter is also studied.

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P4.016 Development and improvement of KSTAR NBI2 ion sources

The KSTAR's NBI heating system achieves an output of about 5.46 MW, 8.7s through the 2017 KSTAR

Campaign. The second NBI system (NBI2) has being installed to increase the output of the NBI heating system. The NBI2 system consists of one on-axis and two off-axis ion sources, and the maximum output is designed to be 6 MW based on the neutral beam output. The ion source of NBI2 is in the same form as the NBI1 ion source except: (1) the electrode terminal size of the source table has been enlarged, (2) the size of the ceramic part has been enlarged to improve the cooling efficiency of the filament feedthrough, (3) the beamlet aperture center of the G3 grid is moved in the outward direction in order to improve the transmission efficiency, (4) the G1 grid of the NBI2 ion source accelerator has been changed the bonding between the body and the manifold to prevent the problem of thermal deformation and clogging of the inner cooling channel which has been a problem continuously. To verify the NBI2 ion source produced, we will carry out the inspections and performance tests: (1) Thermal load test for G1 grid, (2) Thermal shock test of Bellows's O-ring for grid cooling parts, (3) Hi-pot test in vacuum. Inspections include dimension inspection, leak test, pressure test, insulation test, and assemblability. In this paper, we will describe that inspections and performance results of KSTAR NBI-2 ion source by beam conditioning and the 2018 KSTAR Campaign.

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P4.018 LHCD launcher study in KSTAR

It is planned to upgrade the source power of LHCD system to 4 MW in KSTAR for 2021. The system is divided into four sub-systems, which are microwave sources (klystron), high voltage power supplies, transmission lines, and antenna. The conceptual design of each sub-system has been completed. The base design of antenna for 4 MW power is passive active multijunction (PAM) type launcher on the outboard of a mid-plane. One module (0.5 MW) of antenna system including PAM launcher and RF dividing components was developed for the purpose of preliminary study such as a verification of the fabrication method, RF design proof, test of power limitation, accessibility check, and plasma coupling before full scale construction. PAM launcher and RF dividing components were fabricated and their low level RF characteristics have been measured using network analyzer. High power test and conditioning of them are ongoing in a test chamber to increase the power rating with the aim of application to 2018 KSTAR campaign. Although the basic antenna design is outboard PAM, study on launching position for efficient current drive and relevant antenna design is ongoing, e.g. benefits of high field side and top launching, relevant compact antenna design, directivity improvement, simplification of RF feeding line for mid-plane outboard launcher, etc. In this paper, introduction to conceptual design result of 4 MW KSTAR system, test and conditioning result of prototype PAM, and study on compact LHCD antenna adopting standing wave slotted waveguide array with PAM will be presented.

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P4.027 The current status of the system additional heating of the complex of TSP TRINITI and objectives of modernization for the project of tokamak Ignitor

The paper presents the ICR-heating system of the TSP TRINITI complex: purpose; features of implementation; characteristics; power supply system; physical condition; and also discusses the possibility of upgrading the system of ICR-heating of the TSP TRINITI complex for the Ignitor project.
According to the project of tokamak Ignitor to accelerate the plasma ignition and facilitate the access to high confinement regimes such as the H and I regimes need a system of ICR-heating of the 8 generators with frequencies of 80÷115 MHz (115 MHz: Bt=13T, Ip=11MA full performance scenario; and 80 MHz: Bt=9T, Ip=6-7MA reduced performance scenario), the maximum power of the generator 2 MW at 80 MHz and 1 MW at 115 MHz, the pulse duration to 4s [1].

The RF system was created for ICR-heating of the plasma of the TSP TRINITI at the power level of 2 MW with a frequency <30 MHz in the pre-compression stage of the discharge (Bt=2T, ne=0,5×10^20m-3, τRF=0,1s) to improve the efficiency of adiabatic compression of the plasma. The mechanism of absorption of fast waves at cyclotron frequency of small additive of ions ++He3 was used as the main method of heating D or D-T plasma of the TSP TRINITI. The RF system (2 MW, 20 MHz) was successfully tested at the TSP TRINITI [2].
Technical proposals are being developed to upgrade both the RF system and the power supply of the TSP TRINITI complex in accordance with the current requirements of the tokamak Ignitor project.
References
1. B. Coppi, A. Airoldi, R. Albanese et al., Nucl. Fusion 55 (2015) 053011.
2. Mimov S.V., Kovan I.A., Azizov E.A. et al., Proc. 16tn International Conference on Fusion Energy, Montreal, 1996, IAEA, v.1, p.763, Vienna, 1997.

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P4.028 Development of cesium-free negative hydrogen/deuterium ion source by sheet plasma

Negative ions play an essential role for Neutral Beam Injection (NBI) system of steady state magnetic nuclear fusion. We have development of negative ion sounrce in a cesium-free discharge by the magnetized sheet plasma device, TPD-Sheet IV [1]. Negative ions are formed by volume-production, that is, the dissociative attachment of low energy electrons (Te = 1-2 eV) to highly vibrationally excited molecules. These molecules are attributed to the electron-impact excitation of molecules by high energy electrons (Te > 10 eV) in the plasma column. Negative hydrogen/deuterium ion beams are extracted through the small hole (6mm in diameter) with an extraction voltage VEX of 3 kV at a neutral gas pressure Ps of 0.3 Pa. The maximum negative hydrogen ion beam current density JH- is about 7.0 mA/cm2 at discharge current Id of 50 A. On the other hand, the negative deuterium ion beam current density JH- is obtained about 4.0 mA/cm2 at Id of 80 A.
This study is supported by the LHD Joint Project of the National Institute for Fusion Science (NIFS16KOAR021).

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P4.030 AUG flight simulator development

Fenix, the ASDEX Upgrade (AUG) flight simulator under development, is based on the Plasma Control System Simulation Platform (PCSSP) developed for ITER and the ASTRA transport code. Fenix will give a session leader the possibility to check whether the discharge will meet experimental goals prior to execution. It reads the AUG discharge program and checks if all the parameters and reference waveforms are reasonable during the discharge simulation.

ASTRA serves as a plant (physical model of AUG tokamak) which outputs idealised diagnostics signals (temperature, density, etc.). ASTRA calculates them from particle transport in the plasma core, plasma edge, Scrape of Layer (SOL) and divertor. It also includes particle balance, L-H transition and sawtooth models, and is equipped with the 2D equilibrium reconstruction code SPIDER.

The second component of Fenix is a model of the Discharge Control System (DCS). It simulates DCS controllers and AUG actuators. The DCS model processes data from ASTRA, calculates commands for actuators and sends them back to ASTRA closing the feedback loop. Controllers for coil currents to control plasma current, position and shape are also implemented in the model as well as gas puff valves (divertor and midplane) and the pellet injector to control electron density. The DCS model also simulates external heating actuators such as NBI, ECRH and ICRH.

Currently, Fenix models actuators, diagnostics and controllers. It has methods for reading configuration files and archiving. This article presents modelling of the ASTRA and DCS components, and first results from the simulations of the plasma ramp-up and flattop phases.

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P4.033 Closed-loop simulation with Grad-Shafranov equilibrium evolution for plasma control system development

Plasma control design increasingly depends on fast simulations able to connect to operational plasma control system (PCS) software for iterative development. GSevolve is a nonlinear free-boundary simulation that evolves the Grad-Shafranov equilibrium including current and pressure profiles, and can connect to all versions of the DIII-D PCS operational on devices around the world. Its ability to simulate the evolution of current continuously from open to closed field lines enables GSevolve to simulate full-discharge control from breakdown through plasma termination. GSevolve is compliant with the Plasma Control System Simulation Platform (PCSSP), a Matlab/Simulink-based environment that serves as the primary framework for development of the ITER Plasma Control System (PCS). The code employs a novel response function-updating scheme to achieve faster than real-time simulations of ITER. The plasma evolves linearly until a significant shape change triggers a nonlinear update of the response while enforcing a prescribed accuracy. Old responses are recalled if the plasma changes back to an old shape. The equilibrium is a function of a state vector containing axisymmetric conductor currents and parameters related to parallel and perpendicular currents, which evolves according to an efficient state space circuit formulation. Control design and validation results are presented for simulations of DIII-D and EAST, connected to their corresponding versions of the DIII-D PCS.
This material is based upon work supported by the U.S. Department of Energy under Award Numbers DE-FC02-04ER54698 and DE-SC0010685.

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P4.037 Plasma Boundary Reconstruction in JET by Magnetic Measurements

The reconstruction of the Plasma Boundary (PB) in fusion reactors is a critical task in several applications, including plasma control and several off-line studies of the plasma configurations.
A widely shared definition for PB is the Last Closed Magnetic Surface (LCMS) within the vacuum vessel. In the limits of plasma equilibrium conditions, the PB reconstruction should imply the solution of the coupled electro-mechanical plasma equations. The problem is rather complex also in axisymmetric geometry and the solution requires numerical computations not compatible with the needs of a real time response. The problem is more and more complex in fully 3D geometries; therefore, some simplified purely electromagnetic approaches have been proposed, including the fitting of magnetic measurements or the equivalent plasma currents.
In any case, in 3D geometry, the LCMS reconstruction is very time-consuming because, despite new algorithms have reduced their complexity [1], the 3D Magnetic Surfaces (MSs) tracking is very demanding.
In this paper a new very effective algorithm is proposed. The limiter configurations are treated in 2 steps: (a) a preliminary analysis of the first wall to select the points with a vanishing normal field component; (b) the tracking of all the MSs including one of these points and the LCMS selection. The divertor configurations are identified with an iterative algorithm inspired by the bisection approach: at (k+1)-th iteration, the LCMS is looked for between the larger closed and smaller not closed surfaces at k-th iteration.
In both cases, large benefits are derived from the availability of a parallel computing architecture to

carry out the simultaneous construction of a full set of MSs.
The paper discusses the application of the new methodology to various JET configurations.

[1] A. Chiariello, A. Formisano, F. Ledda, R. Martone, F. Pizzo. (2017). "Analytical representation of 3D magnetic surfaces". IJAEM, 54(4), 627-645.

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P4.040 Robust configuration of the JET real-time protection sequencer

The JET Real-Time Protection Sequencer (RTPS) co-ordinates responses for magnetic and kinetic actuators to protect the ITER-Like Wall from possible melting events and other undesirable scenarios. It allows programmable stop responses per pulse, based on alarms raised by other systems.

The architecture combines a modular run-time application developed using MARTe (Multithreaded Application for Real-Time execution) with the top-level JET supervisory and configuration software, Level-1. Operational experience since 2011 drove a requirement to refactor the system in 2017 [1], moving the maximum degree of functionality from compiled code to configuration data, providing more flexibility, maintainability and verifiability of action(s) to be taken during a pulse.

This paper discusses the features of the architecture that made this clean separation of rule-based logic and real-time signal processing possible and practical, including how functions and interfaces between MARTe and Level-1 are organised. It also explains data management to address development, testing, commissioning and operations, each with individual ownership, responsibility and lifecycles. The core technology enabling this is the Level-1 domain specific language, able to manipulate, validate and load into plant configuration parameter sets. The language also enables implementation of advanced user interfaces, providing operators with the tools to focus on essentials tasks for their area of responsibility. It exemplifies this with recent verification and validation of the refactored protection system: unit/low-level integration tests defined by core developers and integration/behavioural tests defined by JET's Plasma Operations Group, respectively, ensuring robust and consistent behaviour. We show how the wide scope and power of this language has enabled evolution of JET operations efficiently and correctly over decades of operational experience.

[1] P. J. Lomas et al. this conference

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P4.042 Design of real-time feedback system for detachment control in COMPASS

Mitigation of the intense heat flux to the divertor is a crucial issue for safety operation of ITER and next-step devices. The divertor heat fluxes can be significantly reduced by operating detached plasmas, where a huge amount of energy carried by plasma particles is converted into isotropic radiation. In COMPASS-U the power decay length is expected to be small due to high magnetic field, which induces high power fluxes in divertor, so the risk of reaching limits of the plasma-facing components is imminent. To ensure safe operation of the divertor, a real-time control system, which adjusts the degree of detachment by impurity seeding in the divertor region, should be implemented.

After performing a series of gas selection experiments nitrogen occurs to be the optimal impurity. The dependence of the target heat flux decrease on amount of seeded nitrogen was estimated. In COMPASS reaching partially detachment regime is confirmed by measurements of the new divertor probe array, so the new proposed RTFS will be using combined measurements of Langmuir and Ball-pen probes as input, which allows real-time calculation of impacting heat flux. The MARTe system was used to implement the RTFS based on measurement of heat flux at two distinct locations

at the outer target. Results from previous campaigns were used for conscientious estimation of time constants, including the most significant part corresponding to impurity propagation from the seeding location. The most dramatic benefit for increasing RTFS working speed was obtained by decreasing length of the gas path, which was reached by relocating the piezo-valve closer to the vessel.

This work allows to prepare the system, which is a starting point for empirical studies of RTFS based on probe measurements in COMPASS. Design of the RTFS is made such that it could be modified later on in accordance with COMPASS-U needs.

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P4.043 ST40 Data and Control

ST40 is a new high field spherical tokamak built by Tokamak Energy Ltd. (TE). When operating at full power, the main parameters of ST40 will be: R0=0.4–0.6m, A=1.7–2.0, Ipl=2MA, Bt=3T, κ=2.5, and it will have up to 2MW of auxiliary heating power, and a pulse length of 1-2s. The first operational phase of ST40, at limited performance, took place in early 2018, and its results are presented by M. Gryaznevich et al. at this conference. This contribution presents the data flow and control systems of ST40 and discusses plans for their future development.

ST40 is controlled by two systems: Machine Control System (MCS) and Plasma Control System (PCS). MCS consists of a central server and a distributed system of National Instruments cRIO crates, each of which communicates with MCS server using TCP/IP. MCS is responsible for 24/7 monitoring and control of the full plant, including non-timed functions such as glow discharge cleaning and boronization. The 1MHz plant clock and the triggering of an array of diagnostics and other subsystems also fall under its remit. During a pulse, PCS will control the plasma current, position and shape using feedforward and feedback controllers. PCS is written in MATLAB/Simulink, converted to C, and built into an executable. Signals used for plasma control are acquired using simultaneous sampling digitizers by D-TACQ Solutions Ltd. and sent to PCS over a fibre optic link.

All the data acquired during a pulse is stored in an MDSplus database. This allows easy platform independent access to the data using any of the various MDSplus APIs, e.g., in LabView, Python, MATLAB, IDL, C, and Fortran. During pulse preparation, the pulse schedule is stored in MDSplus and accessed by both MCS (running on Windows), and PCS (running on Linux).

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P4.047 Choice of the detectors for light impurities plasma studies at W7-X using C/O Monitor system

C/O monitor system is a specially dedicated spectrometer for Wendelstein 7-X, which is planned to be installed before the second Operational Phase (OP 2). It will be a high throughput and high time resolution system which will be able to monitor the main low-Z impurities in the plasma. It will be fixed at nearly horizontal position for the observation of Lyman-alpha of H-like ions of carbon (3,4 nm), oxygen (1,9 nm), boron (4,9 nm) and nitrogen (2,5 nm).

To measure emission of these elements proper selection of detectors is needed. Several options for CO monitor system at W7-X have been considered. One of them was multistrip gaseous chamber, which is already applied in the spectrometer monitoring oxygen and carbon lines at ASDEX-U experiment. Another possibility was to use the gas electron multiplier (GEM) detector, which technology has been intensively developed through last few years. Microchannel plates in combination with luminescent phosphor and a set of Photodiodes have been also taken into consideration. Mentioned types of detectors were used in similar spectrometers installed on the plasma devices, however this

solution introduces unnecessary complication. In the first phases of W7-X operation, when the hydrogen is used as a working gas, the CCD camera has been proposed. Nevertheless, the spectrometer is designed in a such way, that it will be possible to adapt other detecting system in the latter phase of operation. For this reason, GEM or multistrip-like detectors are considered, especially when the deuterium will be used. In this contribution the differences between detectors and the possibility of their application in the term of their future use in C/O monitor for W7-X will be thoroughly discussed.

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P4.048 A neutron spectrometer based on diamond detector for D-D fusion neutron measurements on KSTAR

Neutron spectrometer based on diamond detector has proposed for the D-D fusion neutron measurements at the KSTAR tokamak in future. Neutron flux monitoring and neutron spectrometry at the KSTAR tokamak as well as at fusion reactors are one of important diagnostics. Diamond has excellent properties in environments such as radiation harsh, high temperature, small size, and so on.
The aim of this work is to optimize a neutron spectrometer for the KSTAR tokamak applications. For the work, performance of fast-neutron spectrometer based on diamond detector has carried out using 2.45 MeV neutrons produced from a deuterium-deuterium neutron generator. Capability results of diamond fast-neutron spectrometer for neutron diagnostics on KSTAR will be presented in the conference.

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P4.050 Polarimetric laser beam diffraction in tokamak plasma

In this paper the method of complex geometrical optics (CGO) is applied to the analysis of polarymetric laser beam propagation and diffraction in tokamak plasma. The paraxial CGO reduces the problem of beam diffraction in inhomogeneous medium to the set of ordinary differential equations, which significantly simplifies calculation as compared with full-wave approach. Numerical simulations for Gaussian beam, propagating in the poloidal plane of the inhomogeneous thermonuclear plasma, reveal differences in beam width and phase front curvature evolution in comparison with homogeneous media. Such additional distortion of the beam must be taken into account in polarimetric system modelling. The presented approach may also be applied in other measurement techniques that use electromagnetic beams, like interferometry or refractometry - with limitations discussed in the text.

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P4.052 Shutter systems for various diagnostics at the Wendelstein 7-X stellarator

Shutter systems for various diagnostics at the Wendelstein 7-X stellarator
K. Höchel, R. Laube, C. Brandt, M. Otte, M. Schülke and the W7-X Team
Max-Planck-Institut für Plasmaphysik, Greifswald, Germany

The stellarator experiment Wendelstein 7-X (W7-X) has completed the first two operational phases. During the last operation phase OP1.2a in 2017 a maximum energy limit of 80 MJ within a 24 s discharge was achieved. However, in future operation phases stationary plasma discharges with a length of up to 30 min and an energy limit of 18 GJ are planned. High energy fluxes from the plasma and high energy stray radiation from the heating systems reach the wall protecting elements and the wall. Therefore, many diagnostics need a dedicated shutter system for protecting sensitive components.
The presentation gives an overview of the shutter systems for three different in-vessel diagnostics: the vacuum magnetic flux surface diagnostic, the bolometer and the soft X-ray multi camera tomography system (XMCTS). In addition the shutter systems have also the function to enable measurements of the signal-to-noise ratio and to compensate for temperature drifts (dark measurements), which are especially expected for the long pulse discharges. The local environment and the individual boundary conditions of these shutter systems resulted in sophisticated design solutions.
The following difficulties have been solved: low outgassing in vacuum, movement of mechanical parts under high vacuum conditions, compatibility with a bake-out temperature of 150℃, resistance against 140 GHz microwave stray radiation, low perturbation of the magnetic field by the metallic components (low permeability) and functionality of the shutter system in high magnetic fields of up to 3 T. Furthermore, the choice of the materials is restricted due to the need of low cobalt content (low activation). All shutter systems have been successfully used in the first operational phases.

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P4.060 Heavy ion beam probe design and operation on the T-10 tokamak

Paper describes recent advances in Heavy Ion Beam Probing (HIBP), a unique diagnostics to measure the core plasma potential in the T-10 tokamak (R = 1.5 m, a = 0.3 m, B_tor = 1.5 - 2.5 T). Fine focused (< 1 cm) and intense (<130 mkA) Tl+ beams with energy up to E = 330 keV, equipped by advanced control and data acquisition system provides the measurements in the wide density interval n_e = (0.3 - 5)x10^19 m-3 and the wide range of plasma current 100 < I_pl < 330 kA in Ohmic and ECR-heated plasmas. The multichannel parallel plate electrostatic energy analyzer with a high temporal (≥ 1 ms) and energy (DE / E < 5x10-5) resolution simultaneously provides data on the mean value of plasma potential phi and its oscillations (by the beam extra energy), plasma density oscillations (by the beam current) and poloidal magnetic B_pol oscillations (by the beam toroidal shift) in each of 5 sample volumes (SVs). When SVs are poloidally shifted, we can determine the poloidal electric field E_pol=(phi1- phi2)/x, x~1 cm, and the electrostatic turbulent particle flux G_E×B~n_e E_pol /B_tor. Cross-phase of density oscillations allows us to find the poloidal phase velocity of perturbations, or the plasma turbulence rotation, and the poloidal mode number m. Time evolution of local values of plasma parameters and/or fragments of radial profiles (as long as 2-6 cm) can be measured in a single shot at Low Field Side. The whole profile in the range (0.25 < rho < 1) is available shot by shot for B_tor ≤ 2.2 T. High gain (107 V/A) preamplifiers with 500 kHz bandwidth allows us to study broadband turbulence and various types of quasicoherent modes including tearing MHD modes and Geodesic Acoustic Modes.

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P4.063 Functional and load testing of stepping-motor hexapod for positioning of H-alpha optical components in the ITER port interspace

For an ITER optical diagnostic, the components located within the port interspace area must be equipped with the proper remotely controlled positioning/alignment mechanisms meeting the va-

riety of functional, environmental and load requirements. For the H-alpha diagnostics the major requirements are: 1) high structural stiffness and positioning stability under the ~100kg weight of Long Focus Spectral Telescope (LF-ST, metal-mirror optical unit of ~0.9m size); 2) high static and dynamic stability under the electromagnetic and vibrational interface loads; 3) low backstroke; 4) functionality in the magnetic field up to ~0.2T; 5) thermal and radiation resistance up to ~120°C and ~10 Mrad respectively, 6) Compact 6DoF design.
The modified stepping motors, implemented in the hexapod holding the LF-ST, have been designed to withstand the radiation and magnetic field loads. More details of hexapod design are described in a separate submission to the conference [1].
In this work, the test plans and preliminary results of major functional and load tests are reported, including the linear and angular stiffness, 6 DoF positioning precision and stability, resistance to vibrational, thermal and magnetic field loads.
[1] A.Yusov et al., "Prototyping of stepping-motor hexapod for 6 DoF positioning of optical components in the ITER port interspace", submitted to SOFT-2018

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P4.064 Prototyping of stepping-motor hexapod for 6 DoF positioning of optical components in the ITER port interspace

For an ITER optical diagnostic, the components located within the port interspace area must be equipped with the proper remotely controlled positioning/alignment mechanisms meeting the variety of functional, environmental and load requirements. For the H-alpha diagnostics the major requirements are: 1) high structural stiffness and positioning stability under the ~100kg weight of Long Focus Spectral Telescope (LF-ST, metal-mirror optical unit of ~0.9m size); 2) high static and dynamic stability under the electromagnetic and vibrational interface loads; 3) low backstroke; 4) functionality in the magnetic field up to ~0.2T; 5) thermal and radiation resistance up to ~120°C and ~10 Mrad respectively, 6) Compact 6DoF design.
The modified stepping motors, implemented in the hexapod holding the LF-ST, have been designed to withstand the radiation and magnetic field loads. Also, the actuator design has been updated after the conducted FEM analysis and preliminary stiffness and stability tests of the commonly used actuator mock-ups. The most flexible elements were eliminated, the gear assemblies and joint were redesigned in order to increase the stiffness above the required 10N/um limit. More details on the test results are presented in a separate submission to the conference [1]
[1] E.Drapiko et al., "Functional and load testing of stepping-motor hexapod for positioning of H-alpha optical components in the ITER port interspace", submitted to SOFT-2018

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P4.067 Modification of neutron emission spectrum by Alfven eigenmodes in a deuterium-tritium plasma

Understanding of resonant interaction between particles and Alfven eigenmodes is one of the most important issues for fusion plasma research. Destabilization of modes by energetic ion population and energetic ion redistribution by unstable modes have been studied by many researchers. However, the interaction of bulk ions with modes has almost not been investigated. Because some bulk ions also satisfy resonance conditions, unstable Alfven eigenmodes can create energetic ion tails in fuel-ion velocity distribution functions in the parallel and antiparallel directions to a magnetic field line. When energetic ion tails are formed in fuel-ion velocity distribution functions, an emission spectrum of fusion-produced neutrons are distorted from the Gaussian distribution. The modification of the neutron emission spectrum may affect neutron wall loading. By detecting the non-Gaussian component in a neutron emission spectrum, the energetic ion tails can indirectly be determined. If

the energetic ion tails are create by Alfven eigenmodes, it is important to clarify the degree of the modification of the neutron spectrum for energetic ion diagnostics by measuring the non-Gaussian neutrons and grasping effects of Alfven eigenmodes on the neutronics in the structural materials. In this study, neutron emission spectra are evaluated for several unstable Alfven eigenmodes assuming an ITER-like deuterium-tritium plasma. It is shown that the neutron emission spectrum is modified in the parallel and antiparallel directions to a magnetic field when mode amplitude is large enough to create energetic ion tails.

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P4.068 Thermography of the FTU toroidal limiter during disruptions

The study of the temporal and spatial characteristics of the disruption heat load on the plasma facing components (PFCs) of a limiter tokamak can be relevant also for divertor machines, where the plasma configuration often degrades to L-mode before the thermal quench.
FTU is an all metal limiter tokamak with AISI 304 stainless steel wall and an inboard full TZM toroidal limiter. FTU being a cryogenic tokamak, the in vessel PFCs are usually at very low temperature, making the detection of their surface temperature during the discharge very difficult. Only during disruptions the thermography of the toroidal limiter, where the plasma usually leans on at the onset of the thermal quench and the plasma is pushed during the current quench, can provide surface temperature and heat loads.
In this paper the analysis of the thermographic images of one out of the 12 toroidal limiter sectors (about 35 x 35 cm2) will be reported. These images have been transferred by an array of Germanium lenses, through a long and narrow (width at the entrance of the vacuum vessel about 8 cm) equatorial port, to a fast thermocamera located at the distance of about 2 meters from the limiter. The preliminary quantitative analysis of the heat loads during different types of disruptions as well as their comparison with those inferred by PFCFLUX code modeling will be presented.
The frequent and sometimes massive presence of eroded or mobilizes dust detected by the thermocamera during disruptions will be discussed as well.

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P4.071 Thermal loads and cooling design for ITER in-port low field side reflectometer diagnostic system

Co-author: Ali Zolfaghari [1]

[1] *PPPL*

The current design for the ITER Low Field Side Reflectometer (LFSR) diagnostic system contains six circular waveguides that function as both launch and receive antennas. The waveguides run from the diagnostic first wall (DFW) to the source/detector with the total length of approximate 40 meters. The front end of LFSR, interfacing with the plasma facing DFW and its supporting diagnostic shielding Modules (DSM), is integrated into equatorial port plug 11 (EPP #11) by the Russian Federal Domestic Agency (RFDA). It is required to function in high temperature, high magnetic field, high nuclear radiation, and high vacuum environment in the port zone.

High thermal loads drive the design of the front end waveguides of LFSR system. During plasma operation, the 14 MeV neutrons from DT fusion reactions penetrate into the port plug and in-port diagnostic components. Diagnostic viewing apertures will result in heat loads penetrating deep into the supporting structures. Meanwhile, the surface heat flux of 0.35 MW/m2 has to be applied on those surfaces affected by a viewing factor consistent with the aperture surface position and

orientation. The results indicate that the in-port LFSR structure and components receive about 80% of the total heat loading onto the diagnostic shield module in which it resides.

Due to the specific viewing aperture layout at the front-end of LFSR, the cooling channel design is challenging. The thermal-hydraulic design iterations are performed using ANSYS CFX, which allows the solution of conjugated heat transfer in solid and liquid parts. The ITER Structural Design Criteria for In-Vessel components (SDC-IC) guides the detailed structural aspects of cooling channels, such as the plasma facing wall thickness, thermal ratcheting damage as well as mitigating the high thermal-induced stress to ensure the structural integrity.

SOFT Track and Session: Components – In-vessel components
Oral or poster preference: Poster

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P4.075 Nuclear analysis of the collective thomson scattering system for iter

The Collective Thomson Scattering (CTS) will be the ITER diagnostic responsible for measuring the alpha-particle velocity distribution. Using mirrors, a 1 MW microwave beam is directed into the plasma via an opening in the plasma-facing wall. The microwaves will scatter off fluctuations in the plasma, and the scattered signal is recorded after transmission through a series of mirrors and waveguides. The system will be implemented in drawer #3 of the Equatorial port plug #12 facing the plasma, leading to several components of the CTS system to be directly exposed to neutron radiation from the plasma which can change the properties of the components and reduce their lifetime. In this work, a neutronics analysis is presented for the CTS system with neutron and gamma-ray spectra and heat loads for the main components of the system. Also presented is a study of the nuclear heat loads in the launcher mirror depending on its material composition, complemented with a finite-element thermal analysis performed with ANSYS Mechanical which shows how the nuclear heat loads translate into operation temperatures. Results for the updated system are presented, considering the new reference model with the adjacent diagnostic systems to the CTS drawer as well as the new shielding structure of the diagnostic. All the presented studies were conducted using the Monte Carlo program MCNP6.

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P4.081 High voltage electrical system of 8.56 GHz CW klystron for electron cyclotron heating on QUEST spherical tokamak

Current start-up experiment and Steady-State Tokamak Operation (SSTO) are performed by Electron Cyclotron Heating (ECH) in QUEST spherical tokamak. SSTO discharge can be maintained longer than 2 hours using continuous 8.2 GHz ECH. For a new ECH for SSTO, 8.56 GHz high power klystron system is in preparation. The klystron can work continuously, and its incident RF power is 250 kW maximum. Therefore, SSTO using 8.56 GHz-ECH is promising to generate higher fluxes of heat and particle than those of 8.2 GHz-ECH plasma. The antenna structure for 8.56 GHz EC wave is a focusing mirror and its beam waist is about 20 cm diameter around EC resonant layer. O- and X-mode can be selected by the polarizer unit.
The high voltage (HV) DC power supply for a cathode of the 8.56 GHz klystron has been installed. It needs to satisfy the specification; maximum voltage Vmax = -54 kV, maximum current Imax = 13 A, and working continuously. The output voltage is adjustable under control by thyristors. Its output current depends on beam current of the klystron as a load. HV must be turned off immediately to protect the klystron when arcing events inside the tube or in the transmission line are detected. IGBT-array in series switches off the HV in 20 μs, where the thyristor-switch is relatively slow. In addition, power receiving of 6.6 kV three-phase AC is controlled by three pairs of a HV switch and

an IGBT stack. The HV switch is connected only when the sequence of the klystron operation is on. IGBT stacks are required to turn off 6.6 kV AC faster than HV switches when errors are received. After following actual load test with the klystron, RF output trial and ECH experiment on QUEST is planned.

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P4.087 High voltage direct current power supplies for electron cyclotron resonance heating at ASDEX Upgrade

The electron cyclotron resonance heating group installs four new gyrotrons to enlarge the heating power at ASDEX Upgrade (Axially Symmetric Divertor Experiment). Gyrotrons are microwave oscillators for additional plasma heating. Therefor two new direct current power supplies are needed. One power supply is able to feed two gyrotrons.

The engineering data for both power supplies are:

- dc voltage control up to 60 kV
- dc current load capacity of 100 A
- dc power of 5 MW
- smoothed dc voltage by 12 pulse rectifier and capacitors
- impact load strength while 10 s

Usually two gyrotrons in maximum continuous wave mode operation need a power supply with 55 kV and 84 A load capacity. That means a dc power of 4.62 MW. The constructed power supplies are conventional thyristor based types and the main parts obtained by the Deutsches Elektronen Synchrotron (DESY) in Hamburg.
The operating principle is the following. A step-down transformer is connected to the 10 kV grid and provides 890 V at the low voltage side. The 890 V part is connected to a star point controller. This controlled three phase bridge rectifier adjusts the root mean square voltage of the following high voltage transformer. An uncontrolled three phase bridge rectifier converts the AC voltage to DC. The DC voltage is smoothed by high voltage capacitors. To get a 12 pulse system with lower ripple voltage, two 6 pulse systems have to be connected. The two 6 pulse systems are connected in series in this case. That requires two AC systems with a phase difference of 30° to each other. To reach this, two different vector groups are used.

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P4.089 Towards the completion of the Power Supply system for Resistive Wall Modes control in JT-60SA

The Power Supply system for Resistive Wall Modes (RWM) control in the JT-60SA experiment is an Italian in-kind contribution to JT-60SA within the Broader Approach Agreement.
A very efficient control of RWM is necessary to access plasma currents with high βN values (3-5) sustained for a hundred seconds, as foreseen in the JT-60SA research plan.
The development of the whole RWM control system (coils, power supplies and control) went through several steps of optimization; the final coil configuration consists in three toroidal rings of six coils placed on the plasma side of the stabilizing plate, inside the vacuum vessel; the coils are individually fed by dedicated power amplifiers, controlled in real time.
A key aspect highlighted by the physics studies was the need of a very fast reaction of the control system to prevent an excessive growth of the plasma instabilities and to limit the power required for

their control. The main requirements for the power amplifiers were a peak current of 300 A, output voltage of 240 V, bandwidth of 3 kHz and a latency shorter than 50 μs.
The fulfillment of such a fast dynamics was addressed via an R&D program which led to an innovative design solution adopting hybrid Silicon-Silicon Carbide (Si-SiC) semiconductor for the power amplifiers and to the development of a new sophisticated inverter on-board control.
The schedule of the procurement was finalized in 2015 and was articulated in a relatively longer phase, for the detailed design, prototype development and testing, in comparison with the manufacturing phase, due to the high level of innovation. The delivery to Japan is expected in September 2018. An overview of the main phases of the development and realization of this advanced system, the first in fusion experiments adopting SiC semiconductors, will be presented.

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P4.097 Design of the JT-60SA cryogenic pipe system

JT-60SA is a tokamak device using superconducting coils to be built in Japan, as a joint international research and development project involving Japan and Europe. One design object of JT-60SA is maximization of the plasma volume and the instrumentation port availability in the condition that several buildings, heating instruments, and diagnostics of JT-60U are reused. The cryogenic pipe system of JT-60SA, which transfers cold helium from the helium refrigerator to cold components such as superconducting coils, thermal shield and diverter cryopumps, has two unique features due to the limited space mentioned above.
One of them is a complicated path. Relatively small 11 valve boxes and 5 coil terminal boxes are separately installed around the tokamak cryostat asymmetrically, instead of a simple large distribution box and a current feeder box.
The other is that several inlets to superconducting coils are placed on upper side of coils, which makes difficult requests for structural analyses because of the higher part of coils, the larger displacement due to the gravity load, the seismic load, vacuuming the cryostat, cooling down the cold components, and the electro-magnetic force by the plasma operation. For example, the cryogenic pipe interface of JT-60SA Toroidal Field Coil is displaced about 8 mm, 11mm and 38 mm toward radius, toroidal, and vertical direction.
In order to determine the design, structural analyses are conducted using ANSYS® software, which is a finite element analysis tool. The analysis result must satisfy that the stress for all pipes and their support structures are lower than the allowable stress of the material and there is no physical interference with any other components as a result of displacement. In this work, the strategy of analysis and the design for JT-60SA cryogenic pipe system are reported.

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P4.098 Tokamak coil operations by energy-conservation converters and distributed energy storage

Tokamak operations requires high-current coils. The systems typically used to supply these coils (self-commutated converters, resistive switching network units) present several drawbacks: high reactive power, harmonics, discontinuous loads, dead times, energy dissipated as heat during breakdown. Some traditional solutions involve large flywheels and power factor compensators. The achievable performances can be optimized by distributing the energy storage (DESS) to each coil and by adopting static devices, as supercapacitors.
The resulting architecture exploits the principle of conservation of energy: the energy in the coil circuit is virtually constant, it is only transformed from an electrostatic to a magnetic form (including the energy exchanged with the coupled coils and with the plasma). Instead of freely oscillating, the energy transfer is managed according to the desired scenario by an energy-conservation converter

(ECC). This can be a thyristor or a IGBT/IGCT bridge. In the latter case, the control is faster and suitable also for the breakdown phase.
After the scenario, the energy requested by the coil goes back to the DESS, ready for a new operation. The perfect conservation is impossible and its damping is due to the plasma resistance, the ECC parasitic components, the vessel passive elements, the connections, the supercapacitor nonidealities, while the losses in the coils are negligible in case of superconductors.
The lost energy is restored by the DESS input power supplies. This can be at any value of active and reactive power, depending on the experiment and grid needs, even to implement long operations. A design for 100 s (DTT) will be presented.
Even if the DESS/ECC costs may exceed those of other solutions, they are compensated by the simplification in the distribution system (substation, cable, transformers), by the energy saved at each operation and by the intrinsic advantages in terms of maintainability, modularity and dynamic performances.

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P4.100 Analysis of the effect of a plasma disruption on the DTT Toroidal Field magnets

The long plasma pulse duration and the large thermal loads expected in the EU DEMO reactor, currently under design, represent a challenge for the power exhaust and ask for a new, robust design of the divertor. For this reason, the design of a satellite fusion experiment, the Divertor Tokamak Test (DTT) facility, is being pursued in Italy. This fully superconductive compact tokamak, which must be very flexible in terms of plasma configurations, will be the test bench of several DEMO-relevant divertor solutions.
Among the different ex-vessel coil sub-systems, the 18 Toroidal Field (TF) magnets, cooled by forced-flow supercritical helium at 4.5 K, will be the closest to the plasma. Therefore, during fast plasma disruptions large heat loads are deposited in the TF coils, so that the available temperature margin could be eroded up to the initiation of quench, followed eventually by a fast discharge of the coils. A quench causes additional heat deposition in the magnets and typically venting of helium gas, so that some time is needed to re-cool the vented He and the TF coil before restoring the normal operation of the machine, with an impact on its availability and operating costs.
A detailed thermal-hydraulic model of the DTT TF coils is developed here using the 4C code. The model is applied to the simulation of a plasma disruption, assessing which is the minimum He mass flow rate to be retained in the design, in order to avoid any quench initiation during such a severe transient. The option for the tokamak control system to trigger a fast current discharge after the disruption is also investigated, to assess if a quench could be induced during the dump and thus giving important feedbacks to the protection system design.

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P4.102 Monitoring and Control of the Magnet System of JT-60SA

The construction of the full-superconducting tokamak JT-60 Super Advanced (JT-60SA) is in progress under the JA-EU broader approach agreement. During cool down, nominal operation and warm-up the thermal shields, the superconducting magnets and their structures, the high temperature superconductor current leads (HTS CL) and the divertor cryo pumps have to be supplied with helium at specific flow rates, pressures and temperatures. The monitoring of temperatures, mass flows, and pressures and the control of the helium flows is performed by a Magnet Controller (MC).
A particular abnormal situation is a quench or a fast discharge of a magnet which would require many hours for recovery to normal operation. In order to avoid fast discharges as far as possible,

the MC uses "safety interlocks" to supervise the magnet coils and the HTS CL. These interlocks react on deviations of temperatures, pressures or mass flow rates from their nominal values and initiate appropriate counteractions. If a state reaches a critical threshold, a normal stop (NS) of a magnet is initiated through the "Supervisory Control System and Data Acquisition System" (SCSDAS) by ramping-down the current in the coils. In case of a fast discharge the MC acts directly on the power supplies with parallel information to the SCSDAS.
The thresholds are derived from simulations of the cryogenic loops using the thermal analysis programs FLOWER, HEATER, and SUPERMAGNET.
The presentation will describe the philosophy and logic of the Magnet Controller and explain the calculation of some thresholds.

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P4.114 Simulation study of evolution of helium bubbles in bulk W with vacancies

Tungsten (W) is considered as a primary candidate material for plasma facing components, which endures high fluence plasma in future fusion reactor. Extremely insoluble helium(He) introduced into W exposed to high fluence He plasma tend to agglomerate into bubbles, which play a significant role in the radiation-induced micro-structural evolution and properties degradation. Therefore, to understand the relevant underlying physics, especially the micro-structural evolution process dominated by He defects interactions, is of fundamental importance to the development of plasma facing materials and components. Because in-situ observation is difficult by present experimental techniques, molecular dynamics (MD) simulations have been widely employed to study the atomic-level evolution process of He bubbles in W.
Atomistic simulations have been employed to investigate the atomic-level evolution process of He bubbles in bulk W. In this paper MD simulations were used to study the effects on the early stage of He bubbles formation with the He concentration up to 1at% and temperature (300-2100 K). The results show that the size of He-V clusters increases with increasing in the irradiation temperature and He concentration.The sizes of He bubbles obey the normal distribution. He defects interactions result in W interstitial atoms and vacancies with diferent types and sizes. The effects of He-induced and preexisting vacancies in W on the He bubbles evolution are compared. Furthermore, molecular statics(MS) simulations were firstly performed to study the growth of nano-sized He bubbles from energetic perspective. It is revealed that the formation energy of He bubbles is strongly related with its size and He/V ratio. These results also provide input parameters to larger scale simulations methods, such as in kinetic Monte Carlo methods and rate theory.

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P4.120 CFETR Integration Design Platform: Workflow for geometry design of divertor target

In future fusion reactor, divertor is a key component where large heat flux from the confined plasma need to be exhausted. In CFETR Integration Design Platform, where the unified environment for both physics and engineering design is provided, a workflow is developed for the design of the geometry of divertor targets. According to the 0-D design and equilibrium of the designed plasma shape, the workflow is driven by geometry parameters of divertor targets including distances between the strike point and X point and incline angles for both inner and outer targets. Edge plasma is then simulated to predict the heat flux onto divertor targets by using B2.5, where neutral fluid model is adopted to speed up the workflow. The inputs for B2.5 are set based on plasma density, effective charge number and power into scrape-off layer (SOL) from 0-D design and the predicted SOL width by scaling law. The response surface of peak heat flux on the driven parameters can be then built,

as well as further optimization can be performed by using OPTIMUS software. The engineering restrictions due to space requirement and minimum intersection angle between B-field line and divertor target can also be introduced in the optimization. Based on the CFETR (R=5.7 m) design, the workflow is demonstrated for a reasonable and effective geometry design of divertor targets.

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P4.126 Deuterium retention on the tungsten-coated divertor tiles of JET ITER-Like Wall in 2015-2016 campaign

Erosion, deposition, and fuel retention on different plasma-facing components (PFC) are critical issues for next-step fusion devices such as ITER and DEMO. Since 2011, JET has been operated with the ITER-like wall (ILW): tungsten (W) in the divertor and beryllium (Be) in the main chamber. So far there have been three experimental campaigns in 2011-2012 (ILW-1), 2013-2014 (ILW-2) and 2015-2016 (ILW-3). After each campaign, poloidal set of divertor and main chamber tiles were removed for post-mortem analyses. Analyses after the first two campaigns [1] have shown that the fuel retention is reduced by factor of 10-20 compared to the earlier JET operation with an all-carbon wall. The strike point distribution in the ILW-3 campaign was quite similar to ILW-2 whereas ILW-3 contained more high power plasma discharges than the earlier campaigns. To study deposition and deuterium (D) retention on the divertor, samples from the W-coated divertor tiles exposed in 2015-2016 were measured with Secondary Ion Mass Spectrometry (SIMS). Deuterium implanted samples were used for calibrating the SIMS measurements.
Comparison of ILW-3 results will be made with ILW-1 and ILW-2 results to investigate whether high power discharges have increased the thicknesses of the co-deposited layers and whether they have had an effect on the D retention due to higher absorbed energies. Similar to the earlier campaigns, the thickest deposits are found from the upper inner divertor tiles. After ILW-2, depletion of D was observed at the surface of the tiles due to the hydrogen plasma experiments at the end of ILW-2 [1], whereas ILW-3 ended with D plasmas and D peak is observed at the surface of the samples. Results for the D retention will be presented and compared to Nuclear Reaction Analysis and Thermal Desorption Spectroscopy measurements of ILW-3 samples.

[1] K. Heinola et al. (2017) Phys. Scr. 2017 014063

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P4.140 Profile Tolerances influence on Cryostat Base Section

The ITER cryostat—the largest stainless steel vacuum pressure chamber ever built which provides the vacuum environment for components operating in the range from 4.5 k to 80 k like ITER vacuum vessel and the superconducting magnets. The Cryostat is divided into four section, of which, Base section is most complex because of its web shaped structure sandwiched between two 60mm thick plates with stringent requirements in manufacturing tolerances. The required profile tolerance is 30(+10/-20) mm at weld locations and 20(+/-10) mm at other locations but during manufacturing, the tolerances are observed to be 60mm (+10/-50) mm at weld location and 50mm (+10/-40) at other locations. This increased profile tolerances are expected to affect the structural behavior of Cryostat. The present paper discusses assessment of these tolerances on Cryostat Base Section using FEM software, Ansys. The increased profile tolerances on Base Section are applied using two different methods [2]. Maximum tolerance value was considered and five cases were identified to the complete effect of increased profile tolerances on Base section. Limit Load method [1] is used to analyze structural impact of these increased tolerances cases on Cryostat Base Section. The impacts of critical results were discussed in the end.
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P4.170 Numerical simulations of 3D magnetohydrodynamic flows for dual-coolant lead lithium blankets

In liquid metal blanket concepts for nuclear fusion reactors which are currently under development, the liquid metal PbLi serves as neutron multiplier, breeder material and shield against high neutron radiation. In helium-cooled or water-cooled blanket designs, the liquid metal may flow only at very small velocity since the entire heat released in the liquid metal is removed by water or helium. In the dual-coolant lead lithium (DCLL) blanket higher liquid metal flow rates are foreseen to remove the heat that is volumetrically released in the breeder, i.e. the liquid metal serves in addition as a coolant. The movement of the electrically conducting fluid in the strong magnetic field confining the fusion plasma leads to strong Lorentz forces, high pressure drop, and substantial modifications of velocity profiles compared with hydrodynamic flows.
In a first series of simulations, pressure-driven forced flow has been considered. Since for general applications the magnetic field is not always perpendicular or parallel to the duct walls, the validation of the used numerical code has been extended to such cases by comparison of data with exact analytical or asymptotic results for applications where the magnetic field may have arbitrary orientation with respect to the duct walls. A 3D numerical model has been derived from the most recent design of the European DCLL concept for a DEMO reactor. The model meshes the entire liquid metal domain formed by radial and poloidal rectangular channels, bends and U-turns, but also the conducting sheets of flow channel inserts used for electrically decoupling the fluid flow from the highly conducting duct walls for the purpose of pressure drop reduction. First results for pressure drop and flow distribution of 3D MHD flows in DCLL blankets are presented.

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P4.145 Neutronics Studies for Horizontal Lower Port Option in DEMO

The conceptual design activity of the Demonstration Fusion Power Reactor (DEMO) is in progress in the Power Plant Physics and Technology (PPPT) programme within the EUROfusion Consortium. In this work neutronics studies, fundamental for the nuclear design of DEMO, are presented for the horizontal lower port, an optional concept that is currently under investigation. Two possible configurations of the horizontal lower ports have been analised: the vacuum pumping port, with the pumps located inside the port, and an empty port designed for remote handling. For both configurations three-dimensional Monte Carlo calculations have been performed with MCNP5 in a DEMO Water Cooled Lithium Lead (WCLL) model based on 2017 reactor configuration to assess the neutron flux inside and around the port and the nuclear heating in sensitive components, such as the toroidal field coil (TFC) conductor, the vacuum pump, the port walls and the port closure plate. Different shielding configurations have been considered, by adding shielding blocks at the horizontal lower port entrance. Single and double wall port walls and closure plates with different thickness have been studied to reduce nuclear loads and neutron flux. The shielding solutions provide sensible mitigation of nuclear loads in lower port area. However, the nuclear heating on the TFC exceed the limit of 50 W/m3, mainly due to the radiation streaming through the large divertor pumping duct caused by the lack of shielding liner in DEMO baseline 2017 model assumed in these analyses. Design optimization strategy with divertor liner, reduction of pumping duct opening and additional shielding in lower port are needed to reduce the nuclear loads on TF coil.

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P4.148 DEMO Breeding Blanket temperature evaluation before remote maintenance operation

In DEMO the Breeding Blanket (BB) segments shall be periodically replaced, at end of life or after failure. Any replacement of a BB segment shall be performed by Remote Handling Equipment (RHE) through the vertical upper port: any active action is executed by RHE that clamps a BB segment in the upper portion using a BB transporter. A driving requirement of DEMO commercial technology for RHE is its maximum operating temperature at the BB-RHE interface during BB replacement: the objective of this study is properly to assess this limit, being now the allowable value at 100℃. This study deals with the Helium Cooled Pebble Bed (HCPB) concept.
The initial configuration is considered i.e. with all the BB segments sector (three outboard and two inboard segments) in place, and with some simplifications, the Finite Element model tries to predict if a cooling air flux in natural convection conditions inside vacuum vessel is adequate or if a cooling air flux with forced convection conditions must be adopted. The decay heat produced by the main blanket components (the BB modules caps and lateral walls, the backwalls, the supporting structures, the backplate manifolds, and so on) have been evaluated beforehand: the values applied as body loads in the thermal analyses have been selected within the set related to one month after shutdown. The resulting temperature distribution in the BB segments shows that a forced convection will be necessary. The obtained results, the analysis assumption and future analysis plan have been briefly discussed.

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P4.156 In-vessel viewing system prototype performance measurements and simulation of measurement quality across the ITER in-vessel components

The In-Vessel Viewing and Metrology System (IVVS) is a fundamental tool for the ITER machine operations, aiming at inspecting plasma facing surfaces of in-vessel components for both damage and erosion, both of which are related to the amount of mobilised dust present in the Vacuum Vessel.

Key design improvements from the on-going IVVS preliminary design have recently been incorporated into a prototype system, using the latest COTS opto-electronic components incorporated in a custom opto-mechanical probe. This report describes the results from testing of that prototype and the development of a simulation tool, parameterised by the measured prototype performance, to predict IVVS performance across the ITER in-vessel components.

The IVVS prototype has a modular, flexible layout to allow testing of multiple different technologies and design options, including both AM and FM Light Detection and Ranging (LIDAR). The prototype shows best performance precisions of ~15 microns at 10 degrees incidence angle, and <50 microns at up to 70 degrees incidence angle, at 4m target distance, far exceeding the ITER requirements.

Prototype measurement noise has been experimentally demonstrated to be dominated by the variation in the return power of the light scattered off the surface. The Bi-direction Reflection Distribution Function (BRDF) of a divertor monoblock protoype was measured, and an analytic microfacet surface reflection model was developed to calculate return power as a function of target incidence angle and distance.

Based on this model, an IVVS simulator was built that allows calculation of the measurement noise across detailed models of the full ITER vacuum vessel. Using this simulator, we show the coverage and measurement performance in high detail over the ITER first wall and divertor, and develop statistical representations of the accuracy of full-vessel measurements using the IVVS.

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P4.157 Improvement of manufacturing jigs design for reduction of welding distortion in Vacuum Vessel PS1 through finite element analysis

One of the principal means of controlling welding distortion of metallic welded components is the use of assembly jigs. The knowledge about the effect of jigs on weld distortion is based in the qualitative, generally neither systematic nor documented experience of each manufacturer. Such knowledge needs to be complemented by specific research for components with the tight tolerances, intricate geometry, complex manufacturing sequence, size and wall thickness of the ITER Vacuum vessel. The vessel, of toroidal shape, will be constructed through the welding of nine 40º sectors, being each one formed by the welding of four poloidal segments. The influence of the assembly jigs, and the results of their design optimization process for the control of welding distortion in 'poloidal segment one' (PS1) have been investigated through finite element analysis. The result of the investigation is materialized in the jig design modification, obtained through iterative distortion analyses, identifying at each step the main distortion problems in the interaction between jig and segment. The conclusions of the research show that jig design is the main contributor to control weld distortion of PS1, when compared to welding sequence. Results and conclusions obtained provide highly valuable information to the manufacturer during the design phase, in terms of jig behavior, jig re-design and distortion control.

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P4.162 Evaluation of conservative and innovative manufacturing routes for gas cooled Test Blanket Module and Breeder Blanket First Walls

Several manufacturing routines were developed in KIT for First Wall components of the Helium Cooled Pebble Bed concept: for the Test Blanket Module of ITER in 2010-2015 and the Breeder Blanket for DEMO since 2014. The overall fabrication strategy consists of two main steps: 1) the manufacturing of a semi-finished plate penetrated by channels, and 2) the forming of the plate with channels into the U-shaped final geometry; the forming procedure of a plate with an internal channel structure was demonstrated in full scale in 2017, the results will be documented in this paper. However, step 1) which comprises the manufacturing of the semi-finished plate with the internal channel structure remains an issue. The plate shall be fully penetrated by parallel channels (30 channels for the TBM, 120 channels for the DEMO BB). The void due to integrated channels amounts to about 1/3 of the plate volume. The hydrodynamic channel diameter is in the order of 15 mm, e.g. square shaped. Length-to-diameter-ratios are about 2500/10. Additionally, channel surfaces shall be shaped according to thermal-hydraulic aspects, a certain surface roughness or macro structures (V-shaped patterns) are desirable. The routine shall be compatible to applied codes and industrial mass production. Also high cost efficiency is required. The paper compares different approaches for manufacturing step 1): a) Electrical Discharge Machining, b) Hot Isostatic Pressing and c) Additive manufacturing applied as innovative continuous process. The approaches are analyzed with regard to aspects like maximum feasible length, precision, suitability for application of heat transfer enhancement structures, effort for code implementation and technology readiness level. Finally, cost and mass production related aspects are addressed. The results are an important contribution for a general blanket technology assessment, e.g. in connection with the Test Blanket Modules in ITER as well as with the DEMO Breeder Blanket.

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P4.165 Experimental study of liquid metal magnetohydrodynamic flows near gaps between flow channel inserts

In dual-coolant lead lithium blankets, foreseen in fusion power plants, the liquid metal PbLi flows at sufficiently large velocity to guarantee a suitable removal of the volumetric heat generated in the fluid. The moving electrically conducting fluid under the influence of the plasma-confining magnetic field induces currents that create strong electromagnetic Lorentz forces and a high magnetohydrodynamic pressure drop. Electrically insulating flow channel inserts (FCI) are foreseen for decoupling electrically the liquid metal flow from the well-conducting walls. This reduces currents and associated Lorentz force that is responsible for the major contribution to pressure drop in the blankets. Due to geometrical and manufacturing restrictions, it could be necessary to insert FCIs in several pieces in the long poloidal channels. This leads to the presence of gaps between the inserts, namely to interruptions of the electrical insulation, thus providing local shortcut for currents.
The present paper focuses on experimental studies of three-dimensional liquid metal magnetohydrodynamic flows near gaps between FCIs with the purpose to demonstrate the benefits of the inserts for pressure drop reduction and to quantify the additional pressure drop near the gap that has been predicted by theoretical analyses. Experiments are performed in the MEKKA laboratory at the Karlsruhe Institute of Technology in a parameter range typical for dual-coolant blankets. As test section a circular pipe is used, in which sandwich-type FCIs are inserted. The flow is analyzed by measuring pressure differences along the pipe and electric potential in the fluid and on the walls.

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P4.166 Tritium breeding capability of water cooled ceramic breeder blanket with different container designs

Water cooled ceramic breeder (WCCB) blanket has been considered as the primary design in Japan. There are many thin cooling pipes in the blanket and internal pressure would be applied to the blanket by the pipe rupture. A number of efforts have been made for keeping pressure resistance of the blanket with box structure. Pressure resistance of the box structure could be reinforced with internal ribs or fillets but tritium breeding ratio (TBR) would decrease due to an increase of structure material inside the blanket. Cylindrical structure is well known for more effective than the box structure from the viewpoint of pressure resistance. This study aimed to clarify tritium capability of the blanket different container designs. Box structure, two cylindrical structures with poloidal or radial axis were considered in this study. The dimensions of the blanket were referred to WCCB test blanket module (TBM) in ITER project. TBR of different blanket designs were evaluated by using Monte-Carlo code MCNP5.14 and nuclear data library for fusion neutronics applications, FENDL-2.1. A ratio of breeder and multiplier, gap size around the blanket were regarded as parameters. The tritium breeding capabilities of the blanket designs were analyzed from the perspective of the neutron economy considering neutron escape, captures in iron, and neutron multiplication. Based on the results, the relation between TBR and the neutron economy in different TBM designs was clarified. Finally, blanket designs that could improve tritium breeding capability were proposed.

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P4.168 Preliminary implementation of EU-DEMO primary heat transfer systems for HCPB breeding blanket option

This work concerns the conceptual design of the pipework and the main equipment of the Primary Heat Transfer System (PHTS) of EU-DEMO fusion power plant. EU-DEMO is considered to be the nearest-term reactor design to follow ITER; it shall be capable of demonstrating production of electricity, operation with a closed fuel-cycle and to be a facilitating machine between ITER and a com-

mercial fusion reactor.
In particular the systems related to the Helium-Cooled Pebble Bed (HCPB) Breeding Blanket (BB) option are considered here. During pulse operation, breeding blanket modules will be the main thermal power source; other In-Vessel components such as Divertor (DIV) and Vacuum Vessel (VV), generating thermal power at lower temperature, will join with BB in the definition of the total reactor power. BB, DIV and VV PHTS remove all the In-Vessel generated power and reject it to Power Conversion System (PCS) through a molten salt Intermediate Heat Transport System (IHTS), which is equipped with an Energy Storage System (ESS) to allow a continuous operation at nearly constant electric output in both pulse and dwell conditions, despite of the lack of the fusion power generation in this latter phase of the DEMO operation.

Given the current EU-DEMO plant schemes, the authors explored possible layout solutions and the main technological constraints. 3D CAD models helped estimating more precisely fluids inventory and pressure drops, also weight and length of the piping system, which have a significant impact on the plant cost estimation.

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P4.171 EU DEMO WCLL BB breeding zone cooling system design: analysis and discussion

The Water-Cooled Lithium-Lead Breeding Blanket is a candidate option for the European DEMO nuclear fusion reactor. The blanket is a key component in charge of ensuring Tritium self-sufficiency, shielding the Vacuum Vessel and removing the heat generated in the tokamak plasma. The last function is fulfilled by the First Wall and Breeding Zone independent cooling systems.
Several layouts of the Breeding Zone coolant system have been investigated in the last years in order to identify the optimum configuration that guarantee Eurofer temperature below the limit (550℃) and good thermal-hydraulic performances (i.e. water outlet temperature 328 ℃). A research activity is conducted to study and to compare three configurations which rely on different arrangement of the stiffening plates (i.e. toroidal-poloidal and radial-poloidal), orientation of the cooling pipes (i.e. horizontal, vertical) and PbLi flow path. The analysis is carried out using a CFD approach, thus a three-dimensional finite volume model of an elementary cell of the outboard segment is developed for each configuration, adopting the commercial ANSYS CFX code. The objective of the activity is to compare the Breeding Zone cooling system layouts, identifying and discussing advantages and key issues from the thermal-hydraulic point of view, also taking into account feedbacks from MHD and neutronics analyses. The research activity aims at laying the groundwork for the finalization of the Water Cooled Lithium Lead blanket design, pointing out relevant thermal-hydraulic aspects.

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P4.174 Lithium isotope enrichment by electrodialysis using solid lithium electrolyte

In thermonuclear fusion reactor, tritium generated by nuclear reaction of lithium isotope with mass number six (6Li) and neutron is used as fuel. To maintain the nuclear reaction in the reactor, it is necessary to concentrate the 6Li isotope, which exists at only about 7.8mol% naturally, to 40–90wt%. The mercury amalgam method is the only practical method, but its environmental burden is large [1,2]. As advanced method, Kunugi et al. reported that it is possible to condense the 6Li by electrodialysis using lithium ceramics [3]. Hoshino has studied another electrodialysis method using a porous polymeric diaphragm impregnated with a liquid electrolyte [4]. In either method, in principle it is possible to increase the 6Li concentration by cascading, but the separation factor of one step is small. Most recently, our research using lithium-ion conducting solid electrolyte membranes discovered that by applying voltage with an intermittent profile, the condensation efficiency of 6Li

can be increased [5]. In the present work, we investigate the influence of application time, cutoff time and voltage value on the isotope enrichment rate. The isotopic enrichment rate is estimated by the lithium isotope concentration analyzed by inductively coupled plasma-mass spectrometry. This research shows the followings. The concentration rate in a short time immediately after the start of voltage application is high, the rate is improved with sufficiently long interruption time, and the magnitude of applied voltage and the direction of voltage application also influence.

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P4.180 Radiation damage and helium accommodation in lithium metatitanate as a ceramic breeder material

Lithium metatitanate (Li2TiO3) is one of the leading candidates for application as a breeder blanket material for the helium cooled pebble bed (HCPB) concept.

During operation, transmutation of lithium contained within the blanket material as a result of neutron capture produces both tritium and helium. For efficient tritium production, high ceramic density is desirable in order to increase the overall lithium atom density per unit volume, thereby increasing the probability of interaction between fusion neutrons and lithium atoms. Ceramic breeder materials must also withstand long-term exposure to high energy neutron irradiation and the high operating temperatures associated with fusion reactors.

While a substantial amount of research has been carried out concerning tritium diffusion and release from candidate ceramic breeders, comparatively little is known about the mechanisms of helium accommodation in these materials or the influence of helium on tritium diffusion.

In this work, Li2TiO3 has been synthesised by solid state synthesis and novel sol-gel synthesis methods. Variation of post-synthesis sintering conditions has been employed to produce ceramics with different microstructural properties. Selected samples have been implanted with heavy ions (Ti+) to induce displacement damage analogous to that caused by fusion neutrons, and helium ions to simulate the production of helium within the ceramic.

The purpose of this work is to identify the influence of ceramic microstructure (e.g. grain size, morphology, porosity) on radiation damage resistance and helium accommodation/gas bubble formation in candidate ceramic breeder materials.

In this contribution we report evidence of structural modifications resulting from ion implantation, and the observed behaviour of implanted helium in Li2TiO3 following characterisation by X-ray diffraction, Raman spectroscopy and transmission electron microscopy.

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P4.182 Simulating gas adsorption processes via kinetic modeling

Gas adsorption processes are widely used in industrial applications including vacuum pumping in fusion reactors (getters and cryogenic pumps). In gas adsorption flows mass transfer occurs coupled with heat transfer. Modeling of such flows must be based on kinetic theory by applying the Boltzmann equation (BE) or kinetic model equations or alternatively the Direct Simulation Monte Carlo

Method (DSMC). The latter approach has been successfully applied in several works to simulate gas adsorption at the ITER cryopumps, while more recently, there has been one attempt to introduce kinetic modeling based on the BGK kinetic model equation to investigate the effect of partial thermal accommodation on the sticking coefficient in half-space adsorption [1]. Here, this work is extended to model more general multidimensional gas flows, based on advanced kinetic model equations including the BE. More specifically, the fully-developed flow of monatomic and polyatomic gases through and above long plates is investigated based on the Shakhov and Holway kinetic models with adsorption and injection at the walls. The differences between the various kinetic models as well as between monatomic and polyatomic gases are identified. A comparison with corresponding results obtained by the BE is performed. Furthermore, a two-dimensional gas flow configuration similar to the one in cryogenic pumps is simulated obtaining the adsorption flux in terms of the sticking and the inflow conditions for various gases including hydrogen isotopes. Comparisons with corresponding DSMC results are performed. Overall, it is demonstrated that kinetic modeling is a reliable computational tool in order to effectively simulate gas adsorption processes.

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P4.184 Thermal-hydraulic and thermo-mechanical simulations of Water-Heavy Liquid Metal interactions towards the DEMO WCLL Breeding Blanket design

Within the framework of the EU-DEMO Breeding Blanket (BB) research activities, the Water-Cooled Lithium Lead (WCLL) BB concept is the only one which adopts pressurized sub-cooled water as coolant and a Heavy Liquid Metal (HLM), namely Pb-15.7Li alloy, as breeder and neutron multiplier. Cooling water, characterized by operative conditions typical of PWR fission reactors (temperature in the range of 295-328 ℃ and pressure of 15.5 MPa), is foreseen to flow inside the cooling circuit housed within the breeder and deputed to remove the heat power therein generated. Hence, as a consequence of possible in-box LOCAs occurring within the breeder zone, water-HLM interactions might take place whose investigation has to be considered pivotal for the reactor safety analysis and mandatory during design activities.
Within this framework, C.R. ENEA-Brasimone has adopted the LIFUS5 facility for the investigation of water-HLM interactions, carrying out several experimental campaigns and starting validation activities of numerical codes in order to predict both the thermal-hydraulic and thermo-mechanical performances of breeding blanket components under such accidental scenario.
Thermal-hydraulic analyses reproducing experimental tests have been carried out by means of the SIMMER-III code, a numerical tool able to investigate effects of water-HLM interactions in terms of fluid-dynamic and thermal behaviour during postulated transient LOCA scenarios. From the thermo-mechanical point of view, a numerical approach based on the ABAQUS Finite Element code has been adopted in order to reproduce the aforementioned experimental tests in term of test section stress and strain fields. Finally, results obtained from numerical simulations have been benchmarked against experimental measurements to validate the numerical models set-up, with the aim of assessing and improving their reliability.
Numerical results obtained together with their benchmark against experimental data are herewith reported and critically discussed.

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P4.189 Multifunctional Yttrium Oxide coatings for DEMO breeding blanket concepts employing Pb-16Li eutectic

The realisation of the breeding blanket system represents one of the crucial points in the design of future generation fusion reactors. At the time being, problems regarding materials compatibility persist. To protect structural materials from the harsh environment of breeding blankets, different materials have been studied as corrosion-resistant, anti-permeation coatings. Among them, Yttrium Oxide (Y2O3) has been proposed, mainly due to its structural and thermodynamic stability (no crystalline transition up to 2350℃ and no reduction in the presence of Lithium). Still, traditional Y2O3 films present some drawbacks, like porosity and inhomogeneity, which are detrimental for their application as efficient barriers. Here, we report on advanced Yttrium Oxide coatings deposited by Pulsed Laser Deposition (PLD) technique. Deposited films appear dense, compact and well adherent, without defects. The adhesion is verified through mechanical stress and thermal fatigue. Morphological and crystalline features are evaluated in the case of as-deposited and annealed samples. A deep characterization is performed by Scanning Electron Microscopy (SEM), Energy Dispersive X-ray (EDX), Fourier Transform Infrared spectroscopy (FTIR) and X-ray Diffraction (XRD). The mechanical properties of PLD-grown Yttria are measured with an innovative approach, combining Nano-Indentation and Brillouin spectroscopy. The ceramic material appears quasi-ductile (stiffness comparable to traditional steel) with metal-like behaviour and a remarkable H/E parameter. The electrical characteristics are verified with impedance tests, showing resistance values similar to the ones found in literature (largely above the requirements for breeding blanket insulating coatings). The chemical compatibility of Y2O3 films is tested in the presence of molten Pb-16Li eutectic, showing no degradation, nor corrosion of the steel substrates. At last, preliminary permeation tests suggest the potential of the coating to act as an excellent H2 permeation barrier. To conclude, in this work we present a novel multifunctional Y2O3 coating as a promising candidate for future fusion nuclear systems.

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P4.190 Development of a water-cooled blanket concept with pressure tightness against in-box-LOCA for Japan's DEMO

A conceptual design of breeding blanket module with pressure tightness (as 17.2 MPa) against in-box LOCA has been carried out, based on a pressurized water-cooled solid breeder blanket. The cooling water for DEMO is operated at the PWR water conditions of 15.5 MPa and 290 °C-325 °C. Since the pressure loss of the cooling water system was 1.2MPa, the design pressure of the coolant is set to be 17.2 MPa with the margin of 0.5 MPa. In this design, the breeding zone of the module is divided into 0.1-m-squared cells with rib structure and has simple interior for mass production using a mixed bed of Li2TiO3 pebbles and Be12Ti ones. In the DEMO blanket concept with pressure tightness, therefore, the self-sufficient production of tritium and a cooling system for power generation are required to sustain any DEMO operation.
In the FEM analysis, a rib with the thickness of 0.015m is needed basically to withstand the design pressure of 17.2 MPa. In the CFD analysis, the outlet coolant temperature and pressure drop are 321 °C and 0.32 MPa for the blanket concept, when the maximum neutron wall load is 1.66MW/m2 and the heat wall load due to radiation from the plasma is 0.5MW/m2. In addition, the overall TBR ≥ 1.05 was required, which achieve the self-sufficient production of tritium to sustain its own operation. It was found that the self-sufficient production of tritium is likely to be satisfied with the blanket radial width thickness of 0.7m or more when the thickness of the rib structure is 0.015m as suggested by a stress analysis on in-box LOCA. Further improvement of TBR is confirmed when deuterated water is used in the primary coolant system for DEMO.

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P4.193 Microstructural evolution in neutron irradiated beryllium pebbles at DEMO relevant temperatures

Beryllium pebbles produced by the rotating electrode method were selected as a reference neutron multiplier material for the DEMO fusion reactor blanket. Recently they were characterized for this application after neutron irradiation at relevant temperatures. Under neutron irradiation at elevated temperatures, beryllium suffers from significant volumetric swelling stimulated by transmutation gases, helium and tritium. Insoluble helium precipitates as gas bubbles. In contrast, an exact location of tritium was uncertain, until our recent studies have revealed that tritium is trapped inside He-bubbles directly. Thus, the extended porosity formed within grains and along grain boundaries (GBs) is a natural trap for tritium. Accumulated inventory of this ⊠-radioactive element bears potential risk of burst T-release during operation and complicates nuclear waste management after blanket end of life. However, GBs and triple junctions (TJs) play essential role as accelerated gas-release pathways and are investigated in detail.
This work is devoted to the cross-correlation study of microstructural changes observed by optical and transmission electron microscopy in pebbles irradiated as unconstrained pebbles at 370-650℃ up to generation of 3600-6000 appm He and 370-650 appm T. We thoroughly characterized size distribution and number density of bubble formed inside grains. Significantly larger bubbles with smaller number density were found along GBs. Analytical studies with EDX revealed formation of Fe-Al-Be precipitates suggesting an essential role of iron in beryllium swelling, as these precipitates are always covered by large number of small bubbles. The role of GBs and TJs is increased due to formation of bubble-denuded zones with a thickness of up to several μm, as all helium and tritium produced within these zones precipitate at GBs increasing size of the gas bubbles grown there.
This study is extremely important for reliable assessment of tritium accumulation within 300 tons of beryllium required for the tritium-breeding blanket of DEMO.

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P4.194 Experimental verification of dosimetry reaction rates of Co Nb Au Bi with d-Li neutrons

Conceptual design activity for Advanced Fusion Neutron Source (A-FNS) is being carried out for neutron irradiation test of fusion DEMO reactor materials. We are to apply multi activation foils for A-FNS neutron monitor system in order to measure the neutron spectrum using an activation method with an unfolding code. It is important to evaluate the dosimetry cross section data above 20 MeV neutron because of a lack of experimental data. Therefore, we measured the dosimetry reaction rates using the activation foils with d-Li neutrons at CYRIC of Tohoku University. The neutrons are generated by the reaction between 40 MeV deuteron and solid Li target of 25 mm in thickness, and its spectrum is similar to that of A-FNS. We measured the dosimetry reaction rates of 59Co(n,3n)57Co, 197Au(n,2n)196Au, 209Bi(n,4n)206Bi, 93Nb(n,2n)92mNb, 59Co(n,p)59Fe and 59Co(n,2n)58Co reactions as functions of distances between the Li target and foils, and angles between the beam line and foils, to verify the reaction rate due to differences of the neutron spectrum. We performed experimental analyses using Monte Carlo code, McDeLicious-11, which is an extension of MCNP5, and the latest nuclear data libraries, FENDL-3.1d, IRDFF-1.05 and FENDL/A-3.0. The calculated reaction rates of 59Co(n,3n)57Co, 197Au(n,2n)196Au, 209Bi(n,4n)206Bi and 93Nb(n,2n)92mNb reactions using IRDFF-1.05 and FENDL/A-3.0 were agreed well with the experimental ones. It was found that the calculated reaction rates of 59Co(n,p)59Fe and 59Co(n,2n)58Co reactions were disagreed with the experimental ones in some angles, and these cross section data should be revised.

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P4.198 Iter sixth poloidal field coil insulation qualification

The Poloidal Field (PF) coils are one of the main sub-systems of the ITER magnets. Fusion for energy (F4E) is responsible for supply five of them(PF2-PF6) as in-kind contributions to ITER project. ITER PF6 coil is being manufactured by the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP) as per the Poloidal Field coils cooperation agreement signed between ASIPP and F4E.
An insulation system for ITER poloidal field 6 (PF6) coil must have sufficiently high mechanical strength to sustain the enormous electromagnetics force during tokamak operation. We developed a glass fiber and polyimide interleaved insulation structure, optimized the vacuum pressure impregnation (VPI) curing method, and finally accomplished the insulation qualification items for ITER PF6. The mechanical performance of the insulation system, include the ultimate tensile strength (UTS), fatigue tensile strength, interlaminar shear strength (ILSS), was demonstrated by flat-plate laminated specimens, the insulation to stainless steel bonding performance was verified by the compression/shear strength at 45°. Now the static and fatigue UTS, ILSS, compression/shear strength and void content have been completed. This paper summaries the whole qualification process and the testing results for PF6 coil insulation.

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P4.201 W-Cr and W-Ti alloys consolidated by field assisted sintering with and without carbon diffusion barrier

Tungsten is a refractory metal with very good thermal properties i.e. high thermal conductivity, high melting point, good high-temperature fatigue strength etc. Therefore, tungsten is of great interest in thermonuclear fusion research, mainly acting as a heat sink and an erosion protection on the first wall panels. However, mechanical properties and oxidation resistance of tungsten are rather poor. This can be overcome by alloying of tungsten, which improves the oxidation properties. Field assisted sintering has proved to be a suitable technology for consolidation of tungsten alloys. Field assisted sintering is a method of powder materials compaction by applying pulsed electric current and high pressures. Powder is compacted in graphite die, usually with graphite foil used to prevent the adhesion. In comparison to conventional methods, fabrication is feasible in shorter times and at lower temperatures. Numerous studies performed on various materials have shown that, even at short sintering time and low temperature, significant carbon contamination occurs. Negative effect on the material properties e.g. mechanical or optical was reported. However, similar study on tungsten and tungsten alloys has not been performed yet, even though such contamination is undesirable as well.
In this study, samples with two different compositions, W-10%Cr-1%Hf and W-7%Ti-2%Hf (wt. %), were prepared by mechanical alloying and field assisted sintering using two different foils. Sintering using graphite foil has shown that carbon contamination is a serious issue for tungsten alloys. In attempt to lower the carbon contamination, sintering was performed using tungsten foil serving as a carbon diffusion barrier. It was found that response of each alloy on foil material is different, however both alloys exhibit significantly lower amount of carbides formed during sintering in tungsten foil.

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P4.196 Improved analysis on corrosion profile of liquid lithium in 304 stainless steel by LIBS

Liquid lithium can cause serious corrosion on the surface of metal structural materials that used in the blanket and first wall of fusion device. Fast and accurate compositional depth profile measurement for the boundary layer of the corroded specimen will reveal the clues for the understanding and evaluation of the liquid lithium corrosion process as well as the involved corrosion mechanism. In this paper, laser induced breakdown spectroscopy (LIBS) technology was used to study the ma-

trix element (Fe, Cr, Ni, Mn) and the corrosion medium Li in the surface layer of 304 stainless steel which was corroded by liquid lithium at 450℃ for 50 days. In order to improve analysis accuracy, LIBS signal acquisition with different time delays was performed. The variation curves of the complete spectrum and the corresponding curve of some element characteristics with the acquisition time were compared. The appropriate time delay was selected to ensure a higher signal intensity and, at the same time, to minimize bremsstrahlung interference. In addition, both of the intensity and area of the characteristic lines are used to characterize the specific elements, and the peak area shows a better performance in stability and repeatability. The experimental results reveal that LIBS can be used to detect various elements in the sample, especially Li, which provides us the trend of the element concentration in the depth direction. The experimental data also reveal that Li penetrates into the surface layer of stainless steel with Fe concentration in the surface layer almost unchanged, Ni lossy, Cr and Mn concentration fluctuated. Our study further demonstrates that spectral measurement by LIBS is an effective method in the field of metal liquid lithium corrosion research.

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P4.206 Neutron field study of p(24)+Be source reaction using the multi-foil activation technique

The accelerator-driven fast neutron sources of broad- and quasi-monoenergetic spectra are operated at the NPI Rez Fast Neutron Facility utilizing the Be(thick) and 7Li(C) target stations and the variable energy proton beam (up to 37 MeV) from the isochronous cyclotron U-120M. The beryllium target station standardly operated with 35 MeV proton beam is used for production of IFMIF-like (International Fusion Material Irradiation Facility) neutron energy spectrum up to 33 MeV with mean energy of 14 MeV (corresponds to the ITER-tokamak neutron source). Recently, the source reaction of p(24)+Be was investigated using the 24 MeV protons on beryllium target, and neutron field in close source-to-sample distance was determined employing the dosimetry foils activation method (set of activation foils contained Al, Au, Fe, Ni, Y, Ti, In, Co, Nb, Lu). Neutron spectrum reconstruction from resulting reaction rates was performed using the modified version of SAND-II unfolding code and neutron cross-sections from the EAF-2010 nuclear data library. Obtained neutron energy spectra were validated against the Monte Carlo MCNPX predictions. Developed p(24)+Be neutron field up to 22 MeV represents a useful tool for intensive irradiation experiments, radiation hardness study of materials, and validation of cross-sections in energy range relevant to fusion neutrons (≥ 12 MeV). Neutron characteristics of p(24)+Be neutron source will be discussed in the article.

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P4.211 OXIDATION BEHAVIOUR OF TUNGSTEN WITH VANADIUM ADDITIONS

W-V alloys have already been considered for their use in the first wall armor components, but no information about their oxidation resistance is available. To guaranty passive safety of the future fusion reactors during a loss of coolant accident and the ingress of air in the fusion vessel, a full characterization of the oxidation behavior of new tungsten based materials is required. After a power shut-down, the heavily irradiated first wall components may reach temperatures of ≈1000 ºC due to neutron decay heat. At those temperatures, tungsten based components may form highly volatile radioactive oxides that could be released to the environment.
In the present study, the oxidation behavior of W-2V and W-4V (wt. %) alloys has been evaluated in air between 600 and 800˚C. Thermogravimetric curves prove that oxidation behavior of pure tungsten at 600˚C is drastically modified by vanadium additions. Mass gain for the W-2V is about 10 % lower than that of W-4V or pure tungsten. At higher temperatures, however, vanadium additions degrade the oxidation resistance of pure tungsten to catastrophic levels, being the negative effect

more pronounced as the vanadium content in the alloy increases. Harmful influence of vanadium is maximum at 700˚C, temperature at which the oxidation rate of W-V alloys becomes, at least, four times faster than that found for pure tungsten while at 800˚C the difference is reduced about two times higher. The accelerated oxidation of W-V alloys is related to the evaporation of vanadium oxides in the course of the oxidation, leading to a porous scale which cannot confer any protection to the alloy.

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P4.214 Transmission Kikuchi diffraction on carbon extraction replicas: a novel high statistics precipitate characterization pathway for Eurofer-97 steel.

Innovation in materials technology goes in parallel with advancements in material characterization techniques. Recent years have seen a large increase in use of transmission Kikuchi diffraction (tKD) to solving complex materials science problems, including nuclear materials and irradiation damage. A lack of high statistics in transmission electron microscopy (TEM) characterization of precipitates in steels has long been a challenge. We have developed an innovative method of combining tkD analysis and carbon extraction replicas, using which thousands of precipitates can be characterized and their phases identified in one single scan. This method was applied to study a Eurofer-97 steel in non-standard metallurgical heat. We show that tKD on replicas is highly suitable to identify precipitates in Eurofer-97 steel and very small precipitates, down to ~5-10 nm, can be easily identified. Scanning TEM X-ray mapping of the same regions of the replica were also performed, which allowed conclusions on the chemical composition of the different types of precipitates identified by tKD to be drawn, with unprecedented statistics. This technique opens new pathways to characterizing irradiated and non-irradiated materials for fusion reactor applications and beyond.

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P4.220 Measurement of Neutron Spectrum using Activation Method in Deuterium Plasma Experiment at LHD

In Large Helical Device (LHD) at National Institute for Fusion Science, deuterium plasma experiment with d (d, n) 3He reaction was performed from March to July 2017. The neutron generated in plasma is a direct evidence of this reaction. In addition, the neutron spectrum measurement will be useful in fusion engineering to be able to estimate the activation quantity of the fusion reactor devices. In this study, we performed the activation experiments using multiple activation foils to obtain a neutron spectrum in the vacuum vessel of LHD.
We employed materials such as indium and silicon, which can be radioactivated above threshold energy in this experiment. Therefore, we are able to estimate not only the contribution of 2.45 MeV neutrons generated from d (d, n) 3 He reactions, but also that of 14 MeV neutrons generated from concomitant t (d, n) 4He reactions. Some activation foils were sealed in a polyethylene capsule and were transported to the vicinity of the outermost magnetic flux surface of the plasma in the vacuum vessel by the pneumatic tube. After the foils were activated, we measured gamma-rays from the targeted reaction by the HPGe detector and calculated the reaction rate from the obtained just pulse counts. The reaction rate of each activation foil was generally consistent with the calculation result by MCNP6. As a result of the unfolding with SAND-II code, we obtained a neutron spectrum in the vacuum vessel of LHD for the first time. Because the accuracy of the low energy region of the spectrum is not good, the improvement of the response function is necessary to confirm the actual

spectrum.

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P4.227 Development and Application of Neutronics Design Software SuperMC for Fusion

Super Multi-functional Calculation Program for Nuclear Design and Safety Evaluation, SuperMC, developed by FDS Team in China, is a large-scale integrated software system. Taking neutron transport calculation as the core, SuperMC supports the whole process neutronics calculation containing depletion, radiation source term/dose/biohazard, material activation and transmutation. Besides, SuperMC has three main advanced capabilities including CAD/image-based accurate modeling, intelligent data analysis based on multi-D/multi-style visualization and network collaborative nuclear analysis on cloud computing platform.
The whole process of fusion neutronics design, i.e. radiation transport, depletion, activation, and dose calculations are inner-coupled in SuperMC. Advanced radiation transport methods, such as Global weight window generator (GWWG), leads to the calculation efficiency for ITER analysis speed up by 637 times. For activation calculation, the exponential transformation Chebyshev rational approximation method (CRAM) was adopted, in which long time step is divided into several short time steps to avoid inaccurate results caused by long time step problem. Shutdown dose rate (SDR) calculation based on both rigorous two step (R2S) and direct one step (D1S) methods were developed, in which CAD-based accuracy homogenization method was employed for multi-material activation mesh, and the flexible sector cylinder activation mesh division method was implemented to support direction angle sampling.
Complex irregular geometry can be accurately described by hybrid CSG and facet geometry representation, which enables direct use of complicate CAD models including spline surfaces without pre-processing. Meanwhile physical modeling including materials, sources, tallies, etc. has been developed, by which the complex fusion plasma sources could be modeling by one-click, greatly simplified the fusion radiation safety simulation.
SuperMC has been verified and validated by more than 2000 benchmark models and experiments, including SINBAD, fusion reactors (ITER benchmark, ITER C-lite, FDS-II) and ITER SDR benchmark. It has been applied in over 30 major nuclear engineering projects, such as ITER, Europe DEMO, etc.

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P4.228 Assessment of Serpent 2 application for fusion radiation transport analysis

Radiation transport models for fusion neutronics analysis are becoming increasingly complex, further exacerbating problems in the creation and integration of neutronics models found in traditional analysis methods using MCNP. Serpent 2, an alternative radiation transport code developed at VTT Technical Research Centre of Finland, is considered as a potential method for neutronics analysis in fusion relevant problems. Advantages of Serpent 2 include a more robust universe implementation with capability for the use of a mixed geometry definition with of a combination of constructive solid geometry and mesh-based geometry, making it an excellent candidate as an alternative transport code for fusion neutronics, particularly in the case of complex models. Therefore, investigation into the use of Serpent 2 for fusion relevant neutronics analysis is ongoing with important computational and experimental benchmarking.
In this paper computational Serpent 2 results are compared with those from MCNP6. A spherical model is employed for a basic benchmark of the fusion application of Serpent 2 and its corresponding nuclear data library. Comparison of neutron/photon flux and heating for a fusion DEMO [demon-

stration reactor] model using Serpent 2 and MCNP6 will also be presented.
Preliminary results have shown good agreement with results within statistical uncertainties compared to MCNP for neutron flux in simple spherical models, and comparisons on ITER-like geometry within 1% for neutron heating in blankets. Initial comparisons and investigation show significant advantages in the use of Serpent 2 in fusion neutronics analysis methods with further work required to benchmark against experimental data.

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P4.230 JET work effort data collection for ITER operational radiation exposure optimization.

The occupational radiation exposure (ORE) assessment is one of the key aspects of the licensing process for International Tokamak Experimental Reactor (ITER) currently under construction in Cadarache (France). As this machine is the first of its kind, the maintenance activities for the replacement and repair of components are foreseen to be frequent and complex. In this context the remote handling (RH) equipment will play the main role in reducing the radiation exposure to operators to the high radiation field inside the tokamak building. However, hands-on maintenance will be unavoidable, at least to prepare the areas in which the RH activity is planned to be used.
The first step to building an ORE assessment is to identify the work effort (WE), the time necessary to perform a task multiplied by the number of workers engaged in the task under the predicted conditions.
WE data was collected during the Joint European Torus (JET) maintenance shutdowns over the last 3 years under the framework of the Eurofusion WPJET3 program in order to build a validated WE data base (WE-DB) essential for the ITER ORE analyses. The first versions of the WE-DB dated back to the ORE studies for ITER-FEAT (ITER-Fusion Energy Advanced Tokamak), but at that time they included data based on engineering judgement that needed to be confirmed and validated from the WE data gathered in the fieldwork.
The methodology of the WE collection will be described together with the parameters for updating of the WE-DB based on the monitored activities at JET. In addition, a comparison with the WE data used in the previous ITER ORE assessments will allow optimization of the WE for the tasks traced but will also allow the application of the ALARA process by means of a fast and simple tool.

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P4.233 Benchmark of in-vessel Loss-Of-Coolant-Accident models for an EU DEMO helium-cooled blanket: GETTHEM vs. RELAP5-3D

The EU DEMO reactor, under pre-conceptual design within the EUROfusion Consortium, should produce several MW electrical power from nuclear fusion by the 2050s. DEMO shall be equipped with a Primary Heat Transfer System (PHTS) to remove the thermal power deposited in the plasma-facing components and convert it into electricity, and the associated safety-related components and subsystems require detailed investigations for sizing and safety considerations. Among such subsystems, the Vacuum Vessel Pressure Suppression System (VVPSS) is the system responsible for the mitigation of the in-vessel Loss-Of-Coolant-Accident (in-VV LOCA), consisting in a primary coolant ingress inside the VV due to a break in the FW, divertor or VV. The GEneral Tokamak THErmal-hydraulic Model (GETTHEM) code is being used to analyse the VVPSS behaviour during different in-VV LOCA scenarios. GETTHEM is a fast-running, system-level, transient thermal-hydraulic code developed at PoliTo; although the code is proving useful in performing parametric studies, providing feedbacks to the design teams, the reliability of its predictions should be assessed through a validation process. While the model has been validated against experimental data for an in-VV LOCA

scenario for a water-cooled DEMO BB, here the GETTHEM VVPSS model for a helium-cooled BB is benchmarked against RELAP5-3D, which is widely accepted as a certified nuclear thermal-hydraulic computational tool – including applications to fission power plants. Different scenarios, including the presence of isolation valves in the primary loops, which could help mitigating an in-VV LOCA, are modelled with the two codes. The outcomes of the two tools are compared, in terms of the pressure transient in the VV, with particular reference to the peak value, which has to be kept below the maximum pressure allowed for the VV and which defines the acceptable break sizes for a feasible configuration of connections between VV and expansion volume; the discrepancies are then discussed.

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P4.066 Evaluation of the spectrum unfolding methodology for the neutron activation system of a fusion device

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A Neutron Activation System (NAS) is a highly useful and reliable tool for neutron flux and spectrum measurements in fusion devices such as ITER or ENS. The underlying process involves neutron irradiation of chosen material probes followed by their gamma spectroscopy. The gamma spectra and the further data analyses produce quantitative information on the neutron field characteristics in the irradiation position. In state-of-the-art machines like JET, and for future reactors like ITER, NAS is a principal diagnostic component. For the European test blanket modules (TBM) in ITER, integrated NAS with pneumatic sample transport systems are proposed. The careful analysis of the activation foil measurements is crucial for the successful operation of this system, which forms the subject matter of this paper.
In this work, a fast and efficient methodology is being developed for processing of activation foil data from fusion devices. Starting with the gamma spectra of irradiated probes of selected materials, the method ultimately leads to unfolded spectra of the incident neutrons and uncertainty estimates. The method is tested using data from dedicated experiments at JET, and the Nuclear Physics Institute (NPI) Rez. These measurements utilize few, between 3 and 6 reaction channels for the unfolding. Two spectral adjustment code-packages are adapted for this purpose, and are compared: MAXED and STAYSL. The former is based on the maximum entropy method, whereas the latter follows a least squares algorithm. Monte-Carlo simulations are used to get the guess input spectra required by each code. For the response functions, evaluated data libraries like IRDFF-v.1.05, EAF-2010, ENDF-B/VII.1, and TENDL-2017 are taken as source of dosimetry cross-section data. They are also compared for selection of the most suitable data for unfolding. Finally, a closer investigation is made of the handling of uncertainties in the two codes, and the error propagation in the unfolding procedure.
* See the author list of "X. Litaudon et al 2017 Nucl. Fusion 57 102001"

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Assessment of Critical Heat Flux margin on the Tungsten monoblock design of ITER Divertor

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ITER will operate with a full-tungsten (W) divertor made of 54 water-cooled cassettes equipped with W armoured plasma-facing units. The units that protect the vertical targets are made of W monoblocks bonded to poloidal copper-chromium-zirconium (CuCrZr) cooling tubes.
The maximum design heat flux on W monoblocks located in the vertical target part is specified as 20 MW/m2 for 10 seconds during, so-called, slow transient events.
It is prescribed by the ITER internal components structural design criteria to validate experimentally the thermal-hydraulic design of these units with respect to the occurrence of the CHF event, corresponding to a fast and drastic reduction of the heat transfer capability due to vapour layer formation, and the subsequent failure of the component.
The first part of this paper deals with recent Critical Heat Flux (CHF) experimental results obtained at the ITER Divertor Test Facility (IDTF, St Petersburg, Russian Federation) on Tungsten (W) monoblocks mock-ups, which confirmed the margin to the CHF regards to the design heat flux to be higher than the prescribed value 1.4. Comparison with predictions based on thermal-hydraulics correlations was found satisfactory.
In the second part, the reduction of the CHF margin is assessed, taking account the effect of tilting the vertical targets with respect to the cassette body and the introduction of the 0.5 mm depth toroidal bevel on the monoblocks. In the worst case situation, taking account manufacturing tolerances the reduction of the CHF margin could be in the range of 20%. Taking into account the presence of acceptable bonding defects, the reduction could be as high as 40%.
Meanwhile, the latest experimental results balance the consequences of this reduction so that the CHF margin during ITER operations is confirmed to be 1.4

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

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JT-60SA TF magnet assembly

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JT-60SA will be the world's largest superconducting tokamak when it is assembled in 2020 in Naka, Japan (R=3m, a=1.2m). It is being constructed jointly by institutions in the EU and Japan under the Broader Approach agreement. The assembly of its 400-tonne toroidal field (TF) magnet, designed for an on-axis field of 2.25T, will be completed in summer 2018.

After cryogenic testing in Saclay, France each of the D-shaped TF coils (7.5m high and 4.5m wide) was pre-assembled with its Outer Intercoil Structure (OIS) which supports it against out-of-plane loads. After arrival at the assembly site in Naka, Japan the first 17 coils have been rotated around the previously installed toroidal vacuum vessel and its thermal shield. The final, 18th, coil was inserted together with the final 20° sectors of the vacuum vessel and the thermal shield.

All the mechanical features on each coil casing were machined with respect to the winding pack centreline. Each coil has been placed within approximately 1mm of its nominal position. Customized shims along the inboard straight legs, customized splice plates between adjacent shear panels of the OIS and rotatable offset bushings for the shear pins between coils on the inboard side allow the mechanical interfaces to be adjusted.

The Inconel fasteners between coils were tightened hydraulically as each coil was positioned. After all 18 coils were connected, the weight of the assembly was transferred to kinematic supports equipped with spherical bearings.

This presentation will summarize the design features of the coils that facilitated their accurate assembly and the processes used during the pre-assembly of the OIS, the positioning of the coils in the tokamak and the assembly of the joints between adjacent coils.

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The mega amp spherical tokamak real-time protection system

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The Mega Amp Spherical Tokamak (MAST) is currently being extensively upgraded to provide a system that will be able to add to the knowledge base for ITER as well as testing innovative reactor systems such as the Super-X divertor. MAST-U will have increased coil system and power supply capability. Therefore, to ensure operation of MAST-U within safe engineering limits, the machine protection system was also updated. The protection system is needed to protect against coil faults (such as flashovers) and monitor and calculate, in real-time, quantities such as coil currents, axial coil forces and the stored energy within the coils. Its aim is to be able to react within hundreds of microseconds if there were any inconsistencies that breach pre-defined thresholds. The Real-Time Protection System uses Field Programmable Gate Arrays (FPGAs) which monitors up to 89 different signals collected from 8 different locations around MAST-U at a rate of 0.5MS/s. It filters and calibrates the signals before calculating engineering parameters. The results of these calculations are then compared to thresholds, and events are registered if these are exceeded. These events are then mapped to actions such as disabling specific power supplies or requesting a more gradual stop from the plasma control system. In addition to stop requests, more intricate actions can also be performed, such as sending control parameters to the vertical stability system to shift the position of the plasma in the vessel. This bespoke FPGA-based system enables delivery of deterministic results with high bandwidth and with low latency, all within a highly configurable framework. Here, we will present an overview of system design and its interfaces, the results of the functional testing and commissioning of the system.

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Materials Development for new high heat-flux component mock-ups for DEMO

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Thermo-mechanical stability, oxidation and fuel management are driving issues behind the development of new plasma-facing materials for fusion. In recent years significant progress has been made in developing new material types with enhanced toughness (fibre reinforced tungsten (W) – Wf/W) compared to bulk tungsten, Smart-W, with suppressed oxidation and also new advanced copper based materials with enhanced high temperature strength.
Their properties, such as enhanced fracture toughness at RT (> 20 MPa m0.5), will be discussed. Along with mechanical properties initial tests on erosion, high heat flux (HHF) performance and hydrogen retention are available. HHF tests show that having short fibres at the exposed surface leads to their selective erosion and melting. Initial HHF tests under plasma conditions are ongoing in the linear plasma device PSI-2. Furthermore, from modelling it is shown that including fibres coated with yttria into the tungsten matrix will change the hydrogen uptake as yttria reacts as permeation barriers.
Using these new material for components- requires the development of a new highly integrated design approach – which based on exiting results will be discussed in this contribution. With this in mind we will presented the development of two classes of Wf/W-composite, based on an incorporation of continuous fibres or weaves into a CVD tungsten matrix and Wf/W based on a powder metallurgical (PM) route utilizing short fibres of few mm length.
For HHF components typically two designs are considered – a flat tile design as well as a design similar to mono-blocks. In view of manufacturing full scale HHF components test results regarding first large-scale material samples and mock-ups will be shown. In contrast to typical component designs, this material development is pointing to a divergence from classical HHF components, e.g. incorporating both monolithic and composite materials in one concept.

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TRANSP-based closed-loop simulations of current profile optimal regulation in NSTX-Upgrade

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Active control of the toroidal current density profile is one of the plasma control milestones that the National Spherical Tokamak eXperiment - Upgrade (NSTX-U) program must achieve to realize and sustain high-performance, MHD-stable plasma operation. As a first step towards the realization of this goal, a nonlinear, control-oriented, physics-based model describing the temporal evolution of the current profile has been obtained by combining the magnetic diffusion equation with empirical correlations obtained for the electron density, electron temperature, and non-inductive current drives in NSTX-U [Fusion Eng. Des., 123 (2017) 564–568]. The proposed model has then been embedded into the control design process to synthesize a time-invariant, linear-quadratic-integral, optimal controller capable of regulating the rotational transform profile around a desired target profile while rejecting disturbances. Neutral beam injectors, electron density, and the total plasma current are used as actuators to shape the current profile. The effectiveness of the proposed controller in regulating the rotational transform profile in NSTX-U is demonstrated in this work in closed-loop nonlinear simulations based on the physics-oriented code TRANSP. These high-fidelity closed-loop simulations, which are a critical step before experimental implementation and testing, are enabled by a flexible framework recently developed to perform feedback control design and simulation in TRANSP [Nucl. Fusion 55 (2015) 053033 (15pp)].

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JT-60SA Toroidal Field coils procured by ENEA: a final review

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Within the Broader Approach Program between Europe and Japan for the early realization of fusion with the construction of the JT-60SA tokamak, ENEA, the Italian Agency for New Technologies, Energy and Sustainable Economic Development, was in charge, among others, to supply ten superconducting Toroidal Field (TF) coils for the JT-60SA magnet system. The related procurement started in 2011 and will be concluded in 2018 with the delivery of the tenth coil to the Cold Test Facility in France. The construction of each TF coil module consists of two main phases: manufacturing of the superconducting winding pack and integration in the corresponding casing structure. With all the foreseen modules already completed, this paper intends to provide a critical review of the contract for the manufacturing of the coils highlighting the main difficulties occurred in the different manufacturing steps and evaluating the performances obtained from the measurements taken at the acceptance tests.

O3.A / 213

Overview of Beryllium and Tungsten Dust Studies in JET with the ITER-Like Wall

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Processes of material migration, fuel retention and dust generation are key elements in studies of plasma-facing components (PFC) in the JET tokamak with ITER-Like Wall with beryllium limiters and tungsten divertor. Detailed determination of quantity, location, morphology and size of dust are carried out at JET to respond to the ITER needs for safety assessment and to provide input for modelers. The program comprises analyses of particles deposited directly on various erosion-deposition probes, collected by vacuum cleaning of the divertor and locally sampled from PFC with stickers. Two other strands dealing directly with safety aspects are: (a) deposition of mobile particles on the robotic boom used for all in-vessel operations and (b) the impact of moisture on flaking-off of co-deposits, i.e. dust generation following water leaks. The most important results are:
Total amount of dust collected by vacuum cleaning is about 1-2 g per campaign (19.5 -24.3 h plasma operation), i.e. over 100 times smaller than in JET operated with carbon walls.
Two major categories of Be dust are identified: flakes of co-deposits formed and droplets (2-10 ⊠m in diameter) or splashes originating from molten limiters.
Tungsten dust occurs mainly as flakes from W-coated tiles, while no droplets are found.
The areal density of dust particles sticking to monitors amounts to 300-400 mm-2.
On the boom one detects only very few particles connected with the tokamak operation (e.g. flakes of Be co-deposits), while the robotic operation on PFC results in dust generation.
Water ingress on hot surfaces with co-deposits (200 oC) causes thermo-mechanical stress leading to peeling-off of the layers.
Data have been transferred to modelers and areas of the greatest importance in dust generation and transport have been indicated. Details and consequences for a reactor operation will be discussed and further steps in the research program will be discussed.

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Completion of the JT-60 SA Toroidal Field coils tests in the Cold Test Facility

JT-60SA is a fusion experiment which is jointly constructed by Japan and Europe and which shall contribute to the early realization of fusion energy, by providing support to the operation of ITER, and by addressing key physics issues for ITER and DEMO. In order to achieve these goals, the existing JT-60U experiment will be upgraded to JT-60SA by using superconducting coils. The 18 TF coils of the JT-60SA device and its two spare coils are provided by European industry and tested in a Cold Test Facility (CTF) at CEA Saclay. At the summer 2018, all the 20 TF coils will have been successfully tested at the nominal current of 25.7 kA and at a temperature between 5 K and 7.5 K. The main objective of these tests is to check the TF coils performances and hence mitigate the fabrication risks. These tests allowed checking a certain number of performances of the TF coils: DC/AC insulation, cooling down characterization, RRR of the conductor, pressure drop in the winding pack and temperature margin against a quench. More operation or fault scenarios will even be tested on the two spare coils. This paper will give an overview of the main experimental results obtained during these tests and show some statistics over the 20 coils. The main performances of each coil will be summarized, analyzed and discussed in the light of the expected TF coils performances. The main problems met during tests and solutions found will also be highlighted.

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MARTe based Plasma Position Reflectometry System Integration at COMPASS Tokamak

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Plasma Position Reflectometry (PPR) presents an alternative to the employment of magnetic based diagnostics in the determination of the plasma separatrix position. For future controlled nuclear fusion devices, where harsh radiation environment may induce drifts and even damage magnetic probes, PPR can play a major role as a diagnostic for plasma position control during machine operation.

PPR has been demonstrated at ASDEX-Upgrade, where the Discharge Control System (DCS) framework is used, making it relevant to assess an implementation on a different device and real-time control framework. The COMPASS Tokamak presents suitable conditions for such demonstration and further regular operation, by using the O-Mode reflectometer and the Real-Time Control System (RTCS) based on the Multi-Threaded Application Real-Time executor (MARTe) framework. MARTe is currently under upgrade to a Quality Assurance version, MARTe-QA.

Herein we present the integration of the reflectometer diagnostic on the RTCS, both at hardware and software levels. The MARTe-PPR node runs at a rate matching the COMPASS slow control cycle of 500us at the main control node. Sweep data is acquired during discharges using a 200Msps FPGA based acquisition board. The FPGA firmware was modified to enable RT DMA packet streaming at configurable rates. Streamed data via optical PCIe links is now used to reconstruct density profiles, and to estimate the separatrix position, to be delivered to the main node through UDP. This bidirectional UDP link enables synchronization between the MARTe-PPR and main nodes. Acquisition system, sweep trigger generator and reflectometer are all synchronized using a 10MHz time base clock.

Tests conducted so far have shown that the system performs within the required specifications, enabling the trial of closed loop operation using PPR estimated radial positions during discharges. A migration to MARTe-QA is also discussed.

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Experimental study on MHD effect of liquid metal sheath jet for the liquid metal divertor REVOLVER-D

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A new concept of the liquid metal limiter/divertor, REVLOVER-D, which uses multiple free-falling jets of liquid tin as a target, has been proposed. This concept can accommodate the high heat load of several tens of MW per square meter, whereas formation of the stable continuous flow is one of key issues. To avoid the transformation of the jets into droplets due to surface tension instability, "sheath-jet" concept, i.e., insertion of an internal flow resistance (IFR) (e.g., wires, tapes, and chains) to stabilize the flow, has been considered. In order to investigate the flow characteristics of the liquid metal sheath jet including the influence of the magnetohydrodynamics (MHD) effect, experiments using a liquid metal circulation device has been conducted.
In this experiment, a low melting point alloy, U-alloy78 (Bi57-In17-Sn26 with the melting point 78 degree Celsius) is used as a simulant of tin to reduce the issues in the temperature management and the material compatibility with the flow channel. Free-falling jet of U-alloy78 with a height of ~1 m and a maximum flow rate of ~9 L/min is generated using a magnet pump. The attachment equipped with a two parallel plate magnet has been installed and can provide the magnetic field perpendicular to the flow direction with ~0.2 T. Electric current can be applied to the jet by a direct connection or a TIG-type arc discharge. The tilt of the jet due to Lorentz force has been observed with an electric current of the order of 1 A. Dependence of the flow characteristics on the flow rate, magnitude of the electric current, and the shape or material of the IFR has been investigated.
This work is supported by JSPS KAKENHI Grant Number 16H06140, 16K14530 and the budget of NIFS15UFFF038 of the National Institute for Fusion Science.

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Updated Conceptual Design of the DTT magnet system

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DTT is the acronym of "Divertor Tokamak Test" facility, a project for a compact but flexible tokamak reactor which has been conceived in the framework of the European Fusion Roadmap. It will be built in Italy and shall act as a satellite experimental facility to integrate the extrapolation of the ITER results to the EU-DEMO machine. It is thus mainly aimed at the exploration of different divertor solutions for power and particles exhaust, and to study the plasma-material interaction scaled to long pulse operation. The magnet system conceptual design was originally performed in the 2015, but further investigations lead to the final design presented in this paper. The machine layout and size has been changed: the major plasma radius has been modified from 215 cm of the 2015 design to 208 cm of the present one. The overall magnet system is superconducting and it is based on NbTi and Nb3Sn Cable-in-Conduit Conductors. It consists of 18 Toroidal Field (TF), 6 Poloidal Field (PF) and 6 Central Solenoid (CS) stacked and independently energized module coils. In order to cope with the machine requirements, the Nb3Sn TF coil is characterized by a peak field of 11.8 T on the conductor, operating at 33 kA; the Nb3Sn CS modules are characterized by a peak field of about 13 T, with a conductor operating current of 27 kA; the PF coils are wound using NbTi conductors operating at a maximum peak field of 4.0 T, with operating currents ranging from 20 kA to 30 kA, depending on the PF coil.
In the present work, a general description of the updated Conceptual Design of the DTT magnet system is reported, along with the main outcomes of the analyses and the main design choices which lead to this solution.

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Experimental tritium activities in support to tokamak operation and safety

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[1] *CEA Cadarache*

In the future of fusion activities, the mastering of tritium release is a challenge which needs tritium R&D as well as better understanding of tritium impacts on health and environment. We have been working for years on such topics and the purpose of this presentation is to highlight some of the outcomes of these activities.
First, tritium inventory (TI) of different fusion relevant materials (steel, cement, plastic…) when submitted to tritium absorption will be presented. These results are of high importance for safety studies of room potentially contaminated by a tritium gas release. Large discrepancies of tritium inventory between materials are observed (cement TI ~ 100 Steel TI). Due to surface effects, dust can store much higher quantities of tritium than massive samples (1 µm W particles TI ~500 massive samples TI).
The high tritium inventory in dust and the induced electrostatic charging has a major consequence on particle adhesion and resuspension. We will thus present the consequences of particle charging according to the constituent material of the powders but also of their support: 1) on the transport of the dust in operation and on their contribution to the plasma impurities, 2) on the dust explosion and oxidation processes and the likely releases to the environment during an accident (LOVA or LOCA). Various points concerning the tritiated dust release will be also highlighted: 1) what is the specific metrology for the monitoring of workers occupational exposure? 2) what could be the impact on the health of workers potentially exposed to these tritiated releases? Our answers will rely on results obtained in our laboratory after in vitro exposure of biological samples to tritiated tungsten particles. Waste management of metallic tritiated dust is more complex than for tritiated solid and we will present the technical solutions that we propose to be tested.

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Industry Supported Improved Design of DEMO BoP for HCPB BB Concept with Energy Storage System

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The work described in this paper was performed in the frame of the European Fusion Programme (EUROfusion), Power Plant Physics and Technology (PPPT) section, Balance of Plant (BoP) work package for EU DEMO Fusion Power Plant (FPP). DEMO BoP mainly consists of the Primary Heat Transfer System (PHTS), the Intermediate Heat Transfer System (IHTS) that uses HITEC salt as coolant and is equipped with thermal Energy Storage System (ESS) to cope with DEMO pulsed operation, and the Power Conversion System (PCS). The proposed DEMO BoP configuration is capable of producing electricity continuously through the pulse time (2 hours), as well as dwell time (10 min) of the plant operation. The efficiency of the proposed Rankine cycle of the PCS is obtained to be not less than 39%. While elaborating the improved DEMO BoP design, KIT developed a plant model using the industrial code EBSILON®. The development process of the DEMO BoP model was supported by our industrial partners Siemens Power and Gas Division, and Kraftanlagen Heidelberg (KAH). Improved DEMO plant configuration with IHTS/ESS for DEMO HCPB BB (18 sectors design), including conceptual designs of PHTS, IHTS/ESS and PCS is presented in this paper. The paper also presents detailed specifications of key components of the PCS, including sizing and preliminary cost estimates. So far the industrial components for IHTS/ESS and PHTS are still to be specified, so the foreseen IHTS/ESS and PHTS component designs are based primarily on design calculations performed by the project partners. It is worth to mention that no show stoppers were identified from the industry for the current BOP design. Realization of the real equipment, availability and costs evaluation are foreseen in the next two years. The final design of DEMO BoP might still be different, depending on the further research and development within the Eurofusion Project.

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From manufacture to assembly of the ITER central solenoid

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The Central Solenoid (CS), a key component of the ITER Magnet system, using a 45 kA Nb3Sn conductor, includes six identical coils, called modules, to form a solenoid, enclosed inside a structure providing vertical pre-compression and mechanical support. Procurement of the components of the ITER CS is the responsibility of US ITER, the ITER Domestic Agency of the USA, while the assembly of these components will be carried out by the ITER Organization (IO).

In order to manufacture the CS components, US ITER placed a set of orders with industry. Procurement of all the coil modules was awarded in 2011 to General Atomics, that installed from 2012 to 2014 a dedicated manufacturing line in Poway, CA, enabling manufacture of the first module to start in 2015. Procurement of the structure is split among several manufacturers, using existing equipment, sometimes among the largest ones in the world.

Assembly of the ITER CS will require a dedicated area in the ITER Assembly Hall, conventional tooling and special tooling. US ITER is in charge of the procurement of special tooling, while IO is

responsible for the procurement of the conventional one. A detailed assembly procedure is under development at US ITER, in close collaboration with IO and with the support of CEA. Procurement of the special Assembly Tooling is carried out by US ITER, the main part of the first item, the Assembly Platform, having been manufactured and delivered to IO in 2017.

The paper describes the progress of the manufacture of the CS components and of the Assembly Tooling, provides an overview of the assembly procedure and presents the actions on most critical issues undertaken in preparation of the assembly at the ITER site.

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

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Neutron diffraction measurement of residual stresses in an ITER-like tungsten-monoblock type plasma-facing component

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Neutron diffraction measurements have been carried out for non-destructive characterization of the residual stress fields in a mock-up of the ITER-like divertor target plasma-facing component which consists of 4 tungsten blocks joined to a copper alloy (CuCrZr) cooling pipe via a thick soft copper interlayer. The mock-up was manufactured by the hot radial pressing technique at ENEA-Frascati in the frame of a EUROfusion task (WPDIV 2.1-T001). The neutron diffraction measurements were carried out at FRM II reactor in Garching utilizing the STRESS-SPEC diffractometer at room temperature, with a gauge volume 2x2x2 mm3. Parallel to the mock-up, specimens of stress-relieved W and CuCrZr of the same kind as the armor blocks and the pipe, respectively, were examined to measure initial micro-stress field in the macroscopically un-strained reference state before joining. The 3D stress tensor was determined in two different W-blocks and CuCrZr pipe segments, scanning the mock-up from the outer surface of the W block towards the inner wall of the CuCrZr pipe with the interval of 0.4-0.5 mm. Similar profiles of stress components (axial, hoop and radial) were found in the two investigated mock-up segments. Namely, a tensile residual stress is measured in the CuCrZr pipe side near the bond interface region (up to approximately 300MPa) and a corresponding compressive residual stress in the W block side as expected from the requirement of force balance. Significant magnitude of residual stresses was found in the reference specimens of W and CuCrZr, probably due to incomplete relaxation of the micro-strains during the fabrication process. The results are discussed together with the comparative FEM-based numerical prediction obtained for the same mock-up geometry and fabrication history.

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Thermomechanical simulation of flat tiles mock-ups under high heat flux

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The WEST (W -for tungsten- Environment in Steady-state Tokamak) tokamak, is based on an upgrade of the Tore Supra machine [1]. It consists in implementing an actively cooled tungsten divertor for testing high heat flux technology. Beside the presently tested ITER divertor technology (i.e. W-monoblock), the WEST team has also devoted time, in collaboration with ASIPP (China), into the development of an alternative attractive design for actively cooled W/Cu PFCs capable to sustain

heat loads close to monoblock design. Within this framework, ASIPP has produced two small-scale mock-ups based on the W-armoured flat-tile concept. The mocks-up have been successfully tested in the electron beam facility JUDITH-1 (FZ-Jülich, Germany) under successive thermal cycling at 10, 15 then 20 MW/m², showing the capacity of this technology to remove heat loads over several hundreds of cycles at 20 MW/m2 without obvious indication of damage impacting the heat exhaust capability. However, the post-mortem microstructural analysis of both mock-ups showed some damage (namely, tungsten cracks perpendicular to the incident surface) below the tiles tested at 20 MW/m² [2].
This contribution focuses on thermal structural analysis performed with ANSYS and the associated experimental results. The convective heat transfer coefficients were calculated using the Nukiyama model and the mechanical properties of pure copper at high temperature were extrapolated using the Ramberg-Osgood model. The results, in good accordance with the experiments, are used to explain the defects observed on the mocks up.
The simulation confirms the good thermal performance of this technology, which is easier to manufacture than ITER like monoblocks, and far more resistant to high thermal loads than tungsten coating technology. Even though this technology has a lower critical heat flux than the monoblock concept, it shows a comparable heat exhaust capability and accordingly opens promising prospects for fusion technologies.

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CFETR Integration Design Platform: overview and recent progress

The design of China Fusion Engineering Test Reactor (CFETR) involves complex system structure and intricate constrains, and the design work will be performed by a great number of geographically distributed groups. To support the design work consistently and effectively, the CFETR Integration Design Platform (CIDP) is being developed [IEEE Trans. Plasma Sci. 45 (2017) 512; Fusion Eng. Des. 123 (2017) 87]. Based on an advanced GPU-based design cloud, all designers can work on the CIDP through remote connection to the virtual desktops, meanwhile the data are centralized on CIDP. Three systems, i.e., Data Management System (DMS), Task Management System (TMS) and Integrated Design Framework (IDF), are developed to manage the massive design data, coordinate the design work and provide the design environment. In DMS, the CAD data, together with the documents including design requirements, CAE data, design reports and other critical data, are managed according to the tree structure of plant breakdown structure (PBS). Design change, version control and distribution of the latest design data can be implement efficiently in this system. TMS manages the design tasks for specified purpose generated by top-down decomposition of design requirements. The tasks or sub-tasks can be easily allocated in a tree structure and the designers can be assigned. The approved design data are then linked to and managed in DMS. Besides the management functions, it is also important to provide the unified environment for implementation of detailed design work by IDF. Two sub-framework for physical and engineering design are integrated via a unified graphic user interface, in which modular structure is developed for the physics and component design. The detail contents will be presented in this conference.

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Development and qualification of ITER current lead electrical insulation

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ITER high temperature superconducting current lead is a critical component for the magnet system, which has the benefit of reduction in the heat load of the cryogenic system compare with the con-

ventional current lead. The current lead is located in the coil terminal box and dry box in the ITER feeder system. As a warm to cold transition section, the current lead fed the huge current from the power supply system into the coils.
On the surface of the current lead, one layer composite insulation are made to isolate the high voltage potential to the ground. The main configuration of the current lead is over 2 m long cylinder, but at the cold termination and the cooling inlet, the local geometry is much irregular. So the insulation wrapping and curing technology of the current lead shall be researched and developed to acquire the uniform mechanical and electrical performance.
Now, the multi-stage autoclave curing technology has been qualified in ASIPP, the series ITER current leads are manufacturing based on the qualified procedure. In this paper, the latest insulation progresses on ITER current lead are introduced, the electrical results in the formal qualification are presented and discussed.

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High particle fluence over 10^29 D+m-2 achieved in linear plasma generator Magnum-PSI

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The development of a solution for the removal of heat and particles from the reactor is a key area of present-day fusion research as it determines the performance, lifetime and safety of future fusion power plants. The linear plasma facility Magnum-PSI is capable of exposing materials to steady-state plasma conditions similar to those foreseen in ITER and DEMO. In addition, the machine is capable of reproducing the transient heat and particle loads, with powers up to 1 GWm-2, as they occur during so-called Edge Localized Modes (ELM). This combined plasma and heat loading can be used to study the synergistic damaging effects on a target material. As such, Magnum-PSI is recognized as a key machine to predict plasma facing component (PFC) performance at high particle fluence (> 10^27 D+m-2) and high number of thermal cycles (>10^6 ELM-like events) in the European fusion roadmap [1].
Recently extremely high fluences over 10^29 D+m-2 have been achieved with steady-state power densities over 10 MWm-2. By virtue of the superconducting magnet, fluences over 10^30 D+m-2 are possible within a reasonable time frame, enabling the first lifetime studies of fusion materials in a lab environment. Furthermore, results on high heat load testing of tungsten based divertor plasma facing materials (including hydrogen isotope retention by means of in situ ion beam analysis) and high heat flux components based on liquid metals (including the effect of vapour shielding as protection against transient heat loads) will be presented.

[1] S. Brezinsek et al., Nucl. Fusion, vol. 57, no. 11, 116041, Nov. 2017.

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Multi-scale 3D modelling of a DEMO prototype cable from strand to full-size conductor based on X-ray tomography and image analysis

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The design of a superconducting magnet system of a fusion reactor candidate is usually based on the Cable-in Conduit Conductor (CICC) concept. CICC consist of complex structures with several hundreds of highly packed, multistage twisted superconductor and copper strands, cooling structures

– all wrapped in thin steel foils and jacketed in relatively thick stainless-steel pipes. As recently demonstrated, such complex morphology can be captured by fully 3D X-ray tomography.
One of the three alternative winding pack (WP) options for the toroidal field coils of the European DEMO superconducting magnet system as proposed by the Italian ENEA, features a layer-wound WP design adopting a wind-and-react conductor with high aspect ratio rectangular cross section.
A multi-physics modelling requiring high accuracy from the strand to the full-size conductor scale was tested on two X-ray tomography systems. For the microstructural integrity analysis of the internal tin one mm diameter Nb3Sn strand, manufactured by Western Superconducting Technologies, a microtomograph with ~1 μm feature recognition was used. For the full-size conductor (66x25 mm2 internal dimensions) a high energy (> 300 kVp) microtomograph with voxel resolution of ~20 μm was employed. Voids and/or geometrical imperfections were detected, accurately pin-pointed and locally investigated at high resolution before and after heat treatment. Dimensional measurements on the 3D internal structure of the strand showed the small dimensional changes of the strand after the heat treatment. A significant result was the determination of the twist-pitch factor in a purely non-invasive way. The 3D reconstructions of the CICC cables were used to verify the cabling and compaction processes as well as the structural integrity of the strands, wrapping foils and spiral cooling structures. Also, 3D reconstruction of strand trajectories provided a strong input for a multi-physical modelling/interpretation of mechanical–electrical properties of CICC in cryogenic–electromagnetic environments.

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Safety and environmental studies for a European DEMO design concept

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Pre-conceptual design studies for a European Demonstration Fusion Power Plant (DEMO) have been in progress since 2014. At this stage, while a range of design options are being considered, it is essential that assessments are carried out of the safety and environmental impact of these options. This is not only to ensure that the DEMO plant is optimised for safety performance, but also that it will demonstrate the favourable safety and environmental characteristics of fusion energy as part of its mission.
To this end, safety studies have been under way since the start of the project, to set clear safety objectives and requirements, to analyse the response of the plant to off-normal events, to assess hazardous inventories (principally tritium and neutron activation products) and strategies to minimize them, and to identify the main potential contributors to environmental releases in normal operation and to occupational radiation exposure. Development of codes and models for safety analysis has been accompanied by selected experimental activities focused on improving their validation. Safety analysts have engaged with design teams with the aim of selection of design options to achieve the highest safety performance. At the same time, studies are performed of key aspects of waste management, to minimize the waste burden of DEMO and of fusion power plants that will follow.
In this paper the main achievements of these safety and environmental studies are described, drawing preliminary conclusions and noting safety issues that arise from some aspects of the design choices. The future work needed as the EU DEMO project moves towards the conceptual design stage is also identified.

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An Overview of the EU Breeding Blanket Strategy as Integral Part of the DEMO Design Effort

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[1] *PPPT, EUROfusion PMU*

As an important part of the Roadmap to Fusion Electricity, Europe is conducting a pre-conceptual design study of a DEMO Plant to come in operation around the middle of this century with the main aims to demonstrate the production of few hundred MWs of net electricity and to demonstrate feasibility of operation with a closed-tritium fuel cycle.

This paper provides and overview of the newly revised design and development strategy of the breeding blanket in Europe that has been defined to take into account the input from the DEMO pre-conceptual design activities the findings and recommendations of a thorough technical and programmatic assessment of the breeding blanket program and the EU TBM program, for ITER recently conducted by an independent expert panel [1]. This was conducted to identify, among the available options, the most mature and technically sound breeding blanket concepts to be potentially used as "driver" blanket in DEMO [2] and as "advanced" blanket (to be installed and tested in properly designed segments), having the potential to be more attractive for a First-of-a-Kind (FoaK) reactor, the remaining technical gaps and to align and strengthen the supporting R&D Program. To ensure a coherent and efficient Program, a change of the EU TBM options to be tested in ITER is proposed in order to obtain important and useful information from the two considered breeders (solid and liquid) and the two coolants (helium and water).

[1] M. Gasparotto et al., TBM/DEMO BB Programs Review, Final report September 2017.
[2] i.e., the near-full coverage blanket to be installed by day-1 to achieve electricity production and to achieve tritium self-sufficiency

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Safety aspects on the road towards fusion energy

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On the road toward fusion energy, ITER is the first fusion installation which will have enough radioactive inventory to be potentially dangerous for the public and the environment. As such, ITER has a licensed nuclear facility status and ITER Organization, the operator, has to follow a licensing process through which it has to demonstrate to the regulator that the installation is safe at all stages of its operation. In practice, the operator has to define technical, organizational and human provisions such as to prevent or adequately limit the risks of accidents and the disadvantages (exposure to ionizing radiation, environmental releases and waste) that the installation presents.
All installations which will be created after ITER in order to develop (DEMO…) then make use of fusion energy (PROTO…), will clearly also be installations for which the safety aspects will have to be taken into account. If this is not correctly done, it could be an obstacle (in terms of delay or additional cost) or a stop (no licensing) on the way to fusion energy. Perfectly mastered, it could also be a benefice for fusion power compared with other choices.
This paper begins with describing the safety features of fusion installations (first confinement barrier surrounded by large energy sources, plasma disturbances, explosion hazard…). Then, a state of progress of the ITER safety demonstration is provided: solved issues (the tokamak support design, accident within the neutral beam cell…), issues under IRSN's expertise (Explosion within the vacuum vessel, New Vacuum Vessel Pressure Suppression System…), issues to be solved (Detritiation system efficiency, radioprotection, hot cells, tritium and waste buildings…). Finally, a chapter is dedicated to the possible evolutions of the safety issues for the installations succeeding ITER on the road to fusion energy (decay heat removal, exposure to ionizing radiation, environmental releases…). General comments about dealing with these evolutions close this paper (safety from the earliest design stage, lessons learned, involvement of the regulator…).

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Neutronic challenges on the way from ITER to DEMO

Author: Rosaria Villari[None]

Reliable neutronic assessments are essential for the design and the safe operation of high performance fusion facilities. For the under construction ITER device, accurate and complete evaluations of the nuclear responses from the various radiation sources are mandatory to optimize the shielding design, guarantee a sufficient protection of critical components and minimize the occupational exposure of workers. To reduce uncertainties of the computational predictions and the associated risks for the ITER operations, benchmark experiments are in preparation for the future DT campaign at JET within the EUROfusion Consortium, aimed at validating neutronics tools and data used in ITER nuclear analyses. In parallel, design studies are underway in the EU for the development of a demonstration fusion power plant (DEMO) aiming at the production of electricity and the operation of a closed fuel cycle. The DEMO design and related R&D activities will benefit from the ITER experience. Nevertheless, DEMO should demonstrate full tritium breeding capability to achieve self-sufficiency, which is only partially addressed with ITER through the Test-Blanket-Module (TBM) program. Furthermore, the DEMO operative irradiations conditions will be significantly more demanding and the issues related to the degradation and changes of materials properties, including gas production, activation, erosion and corrosion products, contamination, etc.., and the resulting radiation dose loads, will be more severe than in ITER. The significant in-vessel material damage and activation necessitates the development of proper neutron resistant and low activation structural materials which need to be qualified and tested under intense fusion neutron source. Thus, accurate and reliable nuclear analyses to address DEMO design and safety requirements require specific efforts on the improvement and optimisation of simulation tools and nuclear data supported with dedicated experimental activities.
This paper reviews the main ITER neutronic issues, lessons learnt and implications for the DEMO nuclear design and safety, as well as the current R&D activities on codes and nuclear data development and the supporting experimental program. Further needs and challenges to cover the technological gap between ITER and DEMO and efforts to reduce uncertainty margins are discussed.

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Diagnostics for plasma control - from ITER to DEMO

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The development of the plasma diagnostic and control (D&C) system for a future tokamak demonstration fusion reactor (DEMO) [1] faces significant challenges [2]. These comprise the required reliability of operation, the high accuracy to which the plasma parameters are to be controlled, and the robustness of components and methods against any adverse effects or disturbances.
The ongoing developments for the ITER D&C system represent an important starting point for progressing towards DEMO. ITER diagnostic development is guided by a measurement requirements table, in which different categories of diagnostics for machine protection, basic and advanced control, as well as for evaluation and physics studies are being distinguished. Since ITER is an experiment aiming to explore and optimize in detail the physics of a burning plasma, ambitious targets for space and time resolution as well as for the measurement accuracies have been defined. These are pushing the ITER diagnostic design towards using sophisticated methods and aiming for large coverage and high signal intensities, forcing to mount many front-end components in forward positions. This results in many cases in a rapid aging of diagnostic components, so that additional measures like protection shutters, plasma based mirror cleaning or modular approaches for frequent maintenance and exchange have to be developed.
Under the even stronger fluences of plasma particles, neutron/gamma and radiation loads on DEMO, high reliability and long lifetime of diagnostics can only be achieved by selecting the methods with regard to their robustness, and retracting vulnerable front-end components into protected locations in the machine. Based on this approach, an initial DEMO D&C concept has been elaborated, which is covering all major control issues by main control parameters to be derived using at least two different diagnostic methods. Within this paper, on overview on the current status of D&C development for DEMO will be provided.
References:
[1] G. Federici et al., Fusion Engineering and Design 109-111 (2016) 1464-1474
[2] W. Biel et al., Fusion Engineering and Design 96–97 (2015) 8–15

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A PROs-industry joined effort for the ITER construction: evaluating the impact

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In modern innovation ecosystems, academic and practitioners' studies point out the importance of contracts and relationships between innovation actors. Among others, science-based partners, namely universities and Public Research Organizations (PROs), are valuable external sources of knowledge and fundamental pillars of the innovation ecosystems. Although consensus exists on the fact that linkages between science-based partners and industry are crucial for both parts, measuring their impact is still daunting. In this regard, the following research question arises: What is the impact for industry when collaborating for ITER?
A mixed methodology has been adopted to answer this question. The unit of analysis is the single firm that collaborates in the context of ITER project, also with the involvement of ENEA. A total of 26 Italian contractors and subcontractors were identified and analyzed. Data were collected among secondary sources for each firm (financial report and employment data available on AIDA, patent data available on Orbit database, macro data involving the economic and employment situation of the regions analyzed). At the same time, 6 case studies were conducted with selected firms, and were analyzed by content analysis. Case studies enable to drill down the correlations emerged with statistical analyses, suggest explanations for not significant correlations and provide further insights to detect impacts that cannot be grasped by secondary data.
Descriptive statistics and regression analyses show a positive correlation between the starting of the contract with the return on activities (ROA) and the net financial position. Descriptive statistics show the positive impact of the contracts for the financial performance EBITDA/Sales, not fully confirmed by linear regressions. As well, from general descriptive statistics, the impact on the employment growth appears, but linear regressions do not confirm this evidence. However, case studies support the correlations above, even those not confirmed by linear regressions, and bring into evidence other relevant dimensions of impact: market, innovation, learning, social impact. For SMEs, case studies point out an important strategic impact. The ITER project gives firms awareness of their "know-how" and of the need to stress their competencies and to diversify their businesses in a long-term view.
This contribution provides an overview of a scantly analyzed collaboration: PROs-industry at a firm-level dimension. Policy makers should consider the several dimensions of impact and support firms to transform their huge investment into a long-term value. As well, the study helps managers to be aware of the investments required and set a long-term vision on how to transform them in operative returns. Particularly for SMEs, the award of big ITER contracts implies the change of the vision because it offers the awareness to be competitive and the cash flow to invest also in other businesses.

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Industry Supported Improved Design of DEMO BoP for HCPB BB Concept with Energy Storage System