## Contents

<table>
<thead>
<tr>
<th>Section</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>I1 Invited Talks</td>
<td>1</td>
</tr>
<tr>
<td>I2 Invited Talks</td>
<td>4</td>
</tr>
<tr>
<td>I3 Invited Talks</td>
<td>7</td>
</tr>
<tr>
<td>I4 Invited Talks</td>
<td>10</td>
</tr>
<tr>
<td>I5 Invited Talks</td>
<td>13</td>
</tr>
<tr>
<td>O1 Oral session</td>
<td>18</td>
</tr>
<tr>
<td>O2 Oral session</td>
<td>30</td>
</tr>
<tr>
<td>O3 Oral session</td>
<td>42</td>
</tr>
<tr>
<td>O4 Oral session</td>
<td>54</td>
</tr>
<tr>
<td>O5 Oral session</td>
<td>66</td>
</tr>
<tr>
<td>P1 Poster session</td>
<td>78</td>
</tr>
<tr>
<td>P2 Poster session</td>
<td>271</td>
</tr>
<tr>
<td>P3 Poster session</td>
<td>464</td>
</tr>
<tr>
<td>P4 Poster session</td>
<td>647</td>
</tr>
<tr>
<td>Author index</td>
<td>837</td>
</tr>
</tbody>
</table>
I1.1

The ITER Project: Fusion Technology comes of age

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Established by the signature of the ITER Agreement in November 2006, the ITER project is a critical step in the development of fusion energy: its role is to confirm the feasibility of exploiting magnetic confinement fusion for the production of energy for peaceful purposes by providing an integrated demonstration of the physics and technology required for a fusion power plant. Supported by impressive achievements in fusion technology R&D, manufacturing of ITER components is in full swing across the world and the facility is taking shape at St-Paul-lez-Durance. The international collaboration formed around the production of superconducting magnets for the ITER tokamak has produced over 600 t of Nb3Sn and almost 250 t of NbTi superconducting strand, with 80% of the superconductors required for the ITER magnets now complete, and coil fabrication activities underway in 6 of the 7 partners’ factories. Fabrication of the vacuum vessel and thermal shield is moving forward, while the first elements of the cryostat (~29 m diameter x ~29 m height) have been delivered to the ITER site. R&D prototyping and testing of major elements of systems such as plasma facing components, heating and current drive systems, remote handling and power supplies are also at an advanced stage. With on-site construction of the Assembly Hall and Tokamak nuclear complex advancing rapidly, the organizational framework for the initial installation and assembly activities has been defined. To meet the challenges associated with the management of the manufacturing and construction activities, the overall project organization has been strengthened, with a tighter integration between the ITER Organization Central Team and Domestic Agencies. The presentation will review the progress made in developing the advanced technologies required for ITER, the measures taken to establish a more effective project organization and the status of construction of the ITER facility. Keywords: Fusion, Tokamak, Burning Plasma, Superconductivity, Vacuum Vessel, Cryostat, Nuclear Safety
First operation of Wendelstein 7-X

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The optimized stellarator Wendelstein 7-X (W7-X) has started with the goal to demonstrate steady-state plasma operation at fusion relevant plasma parameters. This is to establish the optimized stellarator as a viable fusion power plant concept. The design of W7-X is based on the optimization of the geometric properties of the magnetic field with the aim to minimize neoclassical transport losses in the collisionless regime, to provide good fast ion confinement in the centre of the plasma, to achieve satisfactory equilibrium and stability properties at high $\beta$, and to demonstrate viable divertor operation. The construction of W7-X took 15 years and was completed in mid 2014. After about one year of commissioning (pump-down, cool-down, magnet ramp-up), the device was ready for operation. The commissioning was successfully concluded by the measurement of the magnetic flux surfaces, which has fully confirmend the basic magnetic field topology. In the end of 2015 the first Helium plasmas were created and soon after, Hydrogen plasma operation was started. A brief overview of the construction and commissioning history of W7-X is given. Initial experiences and first results of Helium and Hydrogen plasma operation are summarized. After only a few weeks of operation, Hydrogen plasmas with $2 \times 10^{19}$ m$^{-3}$ line integrated plasma density, 7.5 keV central electron temperature, 1 keV line-averaged ion temperature could be achieved with only 2.5 MW 140 GHz ECR heating power. The pulse duration of 200 ms is determined by impurities but steadily increases with glow discharge wall conditioning. The main observations on transport (energy, particles, impurities), cyclotron wave heating, density control and plasma-wall interaction are reviewed in this paper. W7-X follows a staged approach to steady-state plasmas. After installation of a water-cooled high heat flux divertor, high power (10 MW) steady-state plasma operation up to pulse lengths of 30 minutes becomes possible.
I1.3

**Status of the JT-60SA Project: an overview on fabrication, assembly and future exploitation**

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JT-60SA is a superconducting tokamak developed under the Satellite Tokamak Programme of the Broader Approach Agreement between EU and Japan, and the Japanese national programme. It is designed to operate in the break-even conditions for long pulse duration (typically 100 s), with a maximum plasma current of 5.5 MA. Its scientific aim is to contribute at early realization of fusion energy, in support to the ITER project and also to future DEMO devices by addressing key engineering and physical issues for advanced plasma operation. The JT-60SA Project has shown steady progress in the last years: from the design of the main components, started in 2007 in a close collaboration between EU and Japan, continuing through the assembly in the torus hall, started in January 2013 with the delivery of the first large European component, the Cryostat Base. Since then big milestones have been achieved, like the complete winding and pre-installation of the three lower Equilibrium Field (EF) coils, the welding of a 340° of the Vacuum Vessel sectors, and the completion of most of the Toroidal Field (TF) Coils. Outside the tokamak hall, large auxiliary plant like the Cryogenic System (CS) and the Quench Protection Circuits (QPC) have been fully installed and commissioned, while the Switching Network Units (SNU) and TF and EF coils Power Supplies (SCMPS) are completing installation on site. Other components such as Cryostat Vessel, Thermal Shields, In Vessel Components and so forth are being manufactured and being delivered to Naka site for installation and commissioning. This paper gives technical progress on fabrication, installation and assembly of tokamak components and ancillary systems, as well as progress of JT-60SA Research Plan being developed jointly by EU and Japanese fusion communities. Keywords: JT-60SA, Superconducting Tokamak, Satellite Tokamak Program, Broader Approach
I2.1
Plasma operation with full W divertor – experiences from JET equipped with the ITER-Like Wall

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⁹ See the Appendix of F Romanelli et al, Proceedings of the 25th IAEA Fusion Energy Conference 2014, Saint Petersburg, Russian Federation

Since installation of the JET ITER-Like Wall more than 30h of plasma operation with the inertial cooled full W divertor took place. Successfully, the divertor plasma-facing components PFCs handled harsh tokamak conditions with (i) high surface temperature excursions passing the ductile-to-brittle temperature and re-crystallisation temperature multiple times, (ii) ITER-relevant steady-state and peak power loads due to more than 1.5 million of transients (edge-localised modes or ELMs), (iii) combined impact of deuterium and intrinsic impurities (C, Be, O) as well as extrinsic impurities like He, Ne, Ar, N2 and Xe, and (vi) multiple complex conditioning cycles with baking, deuterium glow discharges and ion-cyclotron-wall conditioning. Routinely, monitoring discharges have been applied to characterise the impurity content in the plasma and the performance of the tungsten divertor. Overall the bulk divertor showed no impact of damage and only moderate damages of the W-coating CFC tiles could be observed. We present an overview of physics findings obtained from the operation with full tungsten divertor including in-situ observations as well as post-mortem analysis of extracted tiles in different interventions. In particular three aspects will be presented in depth:

Erosion of the full W divertor: The contribution of fast particle and heat burst (ELMs) to the total W erosion of plasma-facing components as well as the fraction of prompt re-deposition are determined. Revealing that in normal operation, the intra-ELM contribution is governing the total sputtering source and determines the lifetime. The subsequent migration of W within the divertor to remote areas has been studies by spectroscopy, deposition probes and post-mortem analysis of dedicated poloidal sectors showing moderate, but unexpected transport to remote areas. Fuel retention: The retained fuel in the divertor PFCs has been determined and classified as implantation and co-deposition with Be resulting in short and long-term retention. The complex dynamic fuel retention behaviour due to surface temperature excursion will be presented and related to modelling descriptions. The role of impurities on the retention will be outlined. Power handling: The passive cooling of the W PFCs required careful operation in order to ensure integrity of the tiles. Dedicated analysis showed that the bulk W divertor with his segmented modules and lamella fulfilled the predicted behaviour. Operation with plasma seeding N2 or Ne leading to divertor cooling which allowed expanding the operational window to ITER-relevant divertor conditions.
I2.2

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 demonstration reactor DEMO. CFETR will be operated in two phases: 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 second phase aims for DEMO validation with a fusion power over 1 GW. Advanced H-mode physics, high magnetic fields, high frequency electron cyclotron resonance heating (230 GHz) & lower hybrid current drive (7.5GHz) together with off-axis negative-ion neutral beam injection will be used for achieving steady-state advanced operation. The detailed design, research and development activities including high field magnet, material, T plant, and physical validation on EAST tokamak aiming high performance steady state operation, future MCF road map will be introduced in this talk.
Design of the COMPASS Upgrade Tokamak

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The COMPASS tokamak with ITER-like plasma shape has been put into operation in 2009 in Institute of Plasma Physics ASCR in Prague. It has been equipped by a comprehensive set of diagnostics for edge and Scrape-Off-Layer (SOL) plasma as well as by a new a system of two Neutral Beam Injectors (NBIs), which enabled to obtain significant results in the field of edge, SOL and divertor physics. In order to enhance the relevance of COMPASS for the future ITER and DEMO relevant studies while benefiting from the knowledge of the team in the field of edge plasma, an upgrade of COMPASS has been proposed. The aim of this upgrade is to use a maximum of the existing infrastructure, while enhancing the parameters of COMPASS mainly in a direction of toroidal magnetic field (up to 5 T) and plasma current (up to 2 MA). To achieve these parameters, new magnetic coils made of copper cooled by nitrogen vapors in order to suppress the Ohmic losses will be used. The corresponding support structure will have to accommodate very high electro-magnetic forces on the magnetic coils as well as on the new vacuum chamber (R = 0.85 m, a = 0.3 m – ITER-like plasma shape). This new vacuum vessel will be equipped by a closed divertor and will enable to generate also the double null magnetic configuration. The power supply system will be modified to provide an energy of approx. 250 MJ and electrical power 100 MW. The plasma will be heated by a flexible heating system, which will consist of five NBIs (4 \times 1 MW and 2 \times 0.5 MW). Such a system will also enable a balanced injection in order to study the plasma stability at low plasma rotation. Under such conditions, the plasma parameters of COMPASS Upgrade in the plasma edge, SOL and divertor regions will be highly relevant for studies necessary for the future exploitation of ITER as well as the design of DEMO.
The magnet system is one of the critical core components of the ITER magnets, defining the machine capabilities to form and drive 15MA 500MW nuclear plasmas for 100s of seconds. The magnets, the largest superconducting magnet system ever built with 50GJ of stored energy, are also technologically highly advanced components using large composite Nb3Sn 4-6K force flow cooled conductors that also, in order to maximise plasma performance and minimise cost, stretch current manufacturing technology to its limits. They work at the highest possible electrical (20-30kV), mechanical (primary stresses up to 600MPa) and superconducting performance consistent with very safe and very reliable operation over the life of the machine. The transition from the design phase to the manufacturing phase of the magnets has required tight integration and occasionally several iterations between the design and the results of manufacturing development. This is particularly the case when qualification tests on as-manufactured components reveal smaller margins than anticipated and the original design or manufacturing route (or both) have to be modified, often leading to knock-on effects on other parts of the magnets or, in a tightly packed machine like ITER, other components (and vice-versa). For example, we have encountered significant issues on Nb3Sn filament breakage under magnetic loads that have had to be resolved (for schedule reasons) by special development and test programs executed within the ongoing manufacturing, leading to the selection of an improved multi-stage cable design. Early issues with the flatness of the radial plates that form the TF coil double pancake ‘units’ were resolved by adjustments in the sequencing of welding and machining steps, achieving a flatness (on a product of dimensions 7mx10mx0.1m) of better than 1mm. The development of current leads using High Temperature Superconductor technology, supported by technology transfer from CERN, required development in brazing, very accurate machining (tolerances of 1/100 mm) and high voltage pre-impregnated insulation to achieve a successful result. The magnets, and the associated feeder system, are now entering the final manufacturing stages. Almost all prototyping and qualification work is completed, the sub-components such as the conductors are nearly fully manufactured (over 80% complete), some of the coil production lines are nearly fully ‘loaded’ and in several cases the ‘first of a kind’ final winding packs are nearing completion. For example, the winding of over half of the 18 TF coils is completed and about one quarter of the total double pancake units have undergone the reaction heat treatment to form the Nb3Sn. Although we can still expect a few manufacturing non-conformities to occur, we can now be confident to detect and correct these without schedule impact. The first component deliveries to the site start in about 18 months and become an avalanche within 3 years: assembling these components, keeping up with the delivery rate is the next challenge. In this paper we review the main manufacturing difficulties that have been overcome and summarise the present production status of the magnets.
I3.2

The IFMIF-DONES Fusion Neutron Source

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Fusion road maps defined by both Europe and Japan, Parties to the Broader Approach Agreement (BA) where the IFMIF/EVEDA project is underway, have yet again confirmed the central need of a neutron source dedicated for fusion materials qualification. In the framework of the BA, engineering design and engineering validation activities are conducted which are targeted to prepare the foundations towards the construction and operation for an International Fusion Materials Irradiation Facility (IFMIF). These design activities have already delivered the Intermediate IFMIF Engineering Design Report (IIEDR), which defines a plant that is suited to qualify materials for advanced fusion reactor concepts including fusion power plants. At the same time, discussions framed within Europe leading to a “Roadmap to the realisation of fusion energy” established that, for the purposes of an early DEMO reactor, requirements for IFMIF could be somewhat reduced, particularly in terms of available neutron flux: the degradation of structural materials, in particular Reduced Activation Ferritic Martensitic steels, would need to be studied in a range of structural damage between 20 and 50 dpa as opposed to the ultimate structural damage range between 100 and 150 dpa. Under these conditions, while maintaining the full energy of 40MeV, the IFMIF beam current could be limited to 125 mA, hence allowing a single accelerator solution. After the completion of the IIEDR design activities in Europe have continued towards the design of a cost-reduced version of IFMIF, so called DONES. The design strategy for DONES has been set to keep the volume comparable to the one of the full IFMIF facility while reaching the upper limit of the reduced testing range still in a reasonable irradiation time of 2 to 4 years. An ad hoc group was established in the frame of the Fusion for Energy governance (GB-TAP) in order to evaluate comparative merits of the various Neutron Source options and DONES was indeed found to be the best option to pursue. In parallel, Japan also confirmed their interest to pursue an almost identical technical solution [1] making it possible to now engage in discussion at the international level to pursue jointly such project. This paper will report on key elements of the DONES design, the expected time and cost frame considered for the facility. Ref. [1]: T. Nishitani et al., Fusion Science and Technology, 66 (2014) pp. 1-8
Methodological approach for DEMO neutronics in the European PPPT programme: Computational tools, data and analyses

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The European Power Plant Physics and Technology (PPPT) programme, organised within the EUROfusion Consortium, aims at developing a conceptual design of a fusion power demonstration plant (DEMO) as a central element of the roadmap to the realisation of fusion energy. Various integrated PPPT projects are being conducted to meet this goal including Breeder Blanket (BB), Safety and Environment (SAE), Magnets (MAG), Materials (MAT), Diagnostic and Control (DC), Divertor (DIV), and Remote Maintenance (RM). Neutronics plays an important role for all of the related activities since it has to provide essential data for the nuclear design of DEMO, assess and verify its performance. This requires, on one hand, the availability of suitable computational tools and data to ensure reliable neutronics simulations of DEMO, and, on the other hand, a coordinated approach for the variety of nuclear analyses performed within the PPPT projects. Accordingly, the PPPT programme builds on a co-ordinated approach for DEMO neutronics including both development works on advanced computational tools and activities related to the nuclear design of DEMO and specific components. A dedicated “transversal” activity was implemented to co-ordinate the neutronics activities in an efficient way across the projects. The consistency of the analyses, e., g., is ensured by a methodological approach specified in guidelines for DEMO nuclear analyses. Development works on the improvement of nuclear data for DEMO are currently conducted in a complementary activity supported by F4E. The paper presents in detail the outlined approach including the development works on advanced simulation tools and their application in PPPT nuclear analyses. The focus is on the methodological approach for DEMO nuclear analyses including blanket design, shielding, activation and radiation dose issues with the discussion of specific examples. In addition, the role of nuclear data for reliable DEMO neutronics design analyses and uncertainty assessments is addressed.
European-WP development on functional materials for diagnostics and H&CD systems in future fusion reactors

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With start of EUROfusion Materials-WP in 2014, functional materials (FM) have been included as a new branch. Their main scopes are issues of optical and dielectric materials for DEMO applications. R&D of these materials are, in particular, essential for Diagnostics and Heating and Current Drive (H&CD) systems that must provide critical services such as machine control, protection, performance evaluation and extensive measurement capability. Therefore, a need was identified to study this complete group of materials with very specific requirements. It includes components and materials related such as, optical components (mirrors, lenses, fibres), insulators and high frequency windows. They will also be subjected to an intense radiation field from the ‘burning’ plasma and, therefore, require radiation resistance. Previous studies mainly focus on ITER conditions but they must survive extended periods in the more hostile environment of DEMO and future power plants. The dose received by FM will be orders of magnitude lower than for plasma facing structural materials, however, their sensitivity to radiation is also much higher (even at doses < $10^{-3}$-3 dpa). Draft designs for the EU-DEMO highlighted several R&D issues requiring evaluation of long-term fluence or dose-related degradation of the required properties, such as aggregation and segregation of radiation-induced-defects and impurities. The workprogramme on FM established in EUROfusion will be presented highlighting the different needs addressed above: (i) the group of present material candidates, for which radiation data exist but not at DEMO relevant fluences; (ii) groups of materials (and respective requirements for components), where relevant information does not exist even at low-medium doses. Long-term research activities on FM have been initiated in parallel with DEMO design activities. The types of materials, aims and radiation tests to be performed are reviewed. In particular, for the diagnostics and H&CD materials, in situ irradiation testing is found to be essential.
A substantial step forward in the realization of the ITER HNB system: the ITER NBI test facility

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The realization of the ITER Neutral Beam Test Facility (NBTF) and the start the experimental phase are important tasks of the fusion roadmap, since the target requirements of injecting to the plasma a beam of Deuterium atoms with a power up to 16.5 MW, at 1MeV of energy and with a pulse length up to 3600s have never been reached together before. The ITER NBTF, called PRIMA (Padova Research on ITER Megavolt Accelerator), is hosted in Padova, Italy; it includes two experiments: MITICA, the full-scale prototype of the ITER injector and SPIDER, the full-size negative ion source. The realization promoted by the ITER organization is carried out with the contribution of the European Union, channeled through the Joint Undertaking for ITER (F4E), of the Consorzio RFX which hosts the Test Facility, the Japanese and the Indian ITER Domestic Agencies (JADA and INDA) and several European laboratories, such as IPP-Garching, KIT-Karlsruhe, CCFE-Culham, CEA-Cadarache. The early start of operation of PRIMA experiments is urgent because sufficient experimental time is necessary to face and solve the issues related to the achievement of the desired performance in time for the ITER operation, requiring NBI since the beginning. Substantial progresses have been recently achieved: the buildings construction, begun in October 2012, has been completed by the end of 2015 and the installation of some components has been started since the end of 2014. The SPIDER realization is well advanced: the installation phase is proceeding in good agreement with the general plan; it is expected to be almost completed by the end of 2016. In parallel, the commissioning of the SPIDER power supply (PS) and auxiliary plant systems is being proceeding. Tests at full power and remote control are planned, including also those addressed to reproduce the grid breakdowns and to test the relevant protections. The design of the MITICA injector components was completed in 2015, their procurement is being made through a number of tenders, some of them already launched. The HV Power Supply system of the MITICA 1MV accelerator, provided by JADA, was delivered on site in December 2015. The challenging installation of these components, including the step-up transformers and the SF6 gas insulated transmission line, started soon after and will go on throughout 2016. The present phase, with the PRIMA buildings continuously filled with new components, with the installation activities progressing and with also the commissioning and testing phase starting represents a substantial step forward toward the main target. The paper will describe the main challenges the Project Team has dealt during this phase and the important feedback derived for the ITER HNB systems both from the technical and the organizational standpoints.
A double success story: the international cooperation to build the new ICRF antennas on ASDEX - Upgrade and the results obtained

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A enhanced impurity production has often accompanied experiments using ICRF (Ion Cyclotron Range of Frequency) as heating method. Positive effects, such as the capability to deposit the power centrally even at high density and thereby reduce the central impurity accumulation, were wiped out in the all-metal ASDEX Upgrade when the antenna limiters were also coated with W. The hypothesis that this enhanced interaction is due to RF sheaths - possibly a consequence of currents induced at undesirable locations - has, up to now, not been invalidated. New 3-strap antennas in ASDEX Upgrade, designed to lower these RF sheaths by cancelling these undesirable currents induced in the antenna frame, have shown experimentally to indeed reduce the impurity production. The antennas were designed and fabricated in an international cooperation between IPP, ASIPP (Hefei, China) and ENEA (Frascati, Italy) at different sites whereas installation and test was done by a multinational team at IPP. We report on the points to pay attention to for such international cooperation, on the tools and approaches we used that helped set up a good cooperation, on the final outcome as well as on those aspect we would tackle differently for a similar situation in the future. The antenna also integrates a powerful reflectometer system, designed and built by ENEA and IST (Portugal) from which the first results are coming in. We elaborate on the concept on the basis of which those antennas were built and present an overview of the results obtained. 1 http://www.euro-fusionscipub.org/mst1
Diagnose for First Operation of the Wendelstein 7-X Stellarator

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The mission of Wendelstein 7-X is to assess the reactor capabilities of the HELIAS stellarator line. W7-X is equipped with superconducting coils (B=2.5 T) and is sufficiently large (V=30 m$^3$) to potentially attain steady-state plasmas at low collisionalities and high densities at the same time. As prerequisite for long-pulse operation, W7-X will employ high power, cw microwave heating (initially >5MW) along with NBI and ICRH. The first operation phase (OP1.1) started late 2015 with five uncooled inboard limiter stripes made of graphite. This very first campaign will be followed by a one year operation phase OP1.2 from 2017 onward, following the exchange of the limiters by 10 uncooled 3D divertor modules, shape-wise exactly mimicking the actively cooled, steady-state-capable high heat flux divertor which will be installed for operation phase 2 (OP2), starting in 2020. The first campaign OP1.1 allowed commissioning and demonstration of the overall device operation, i.e. its primary aim was the demonstration of the proper functioning of the control and safety systems of all main device components, like vacuum system, cryogenics, magnetic field coils, ECRH heating and their interplay. Furthermore, this phase was used for putting into operation a significant fraction of the complete set of diagnostics for the divertor operation phase OP1.2 and for experimentally demonstrating the existence of nested flux surfaces. Assuming the heat loads can be spread out evenly between the limiters, 1 second discharges at 2 MW of heating power could be run in OP1.1. The expected plasma parameters are sufficient to demonstrate the readiness of the installed diagnostics and even to run a first physics program, albeit restricted to relatively short pulses, and limiter configurations. The diagnostics available for this first operation phase, including some special limiter diagnostics, and their capabilities will be presented. This will include engineering challenges in view of steady-state stellarator operation (long pulse operation, provision of machine safety, stray-radiation hardening, and 3D mechanical engineering). First experiences with the operation of the diagnostics will also be addressed. Furthermore, a survey on physics-requirement driven engineering developments (e.g. divertor manipulator) for future campaigns (OP2 and beyond) will be given. This project has received funding from the European Union’s Horizon 2020 research and innovation programme under grant agreement number 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.
I5.2

WEST Project status and Research plan towards ITER risk mitigation


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The WEST project is targeted at minimizing risks for ITER divertor procurement and operation. It consists in implementing an actively cooled tungsten divertor for testing the ITER divertor technology under tokamak conditions in Tore Supra. The present paper gives an overview of the project status, and describes the main lines of the associated research plan. As far as the project is concerned, the first assembly phase has been completed in spring 2015 (dismantling of the existing internal components and installation of the outer protection panels). At present, the installation of the divertor coils is underway. The divertor supporting structure has been manufactured and implemented in the machine. Brazing of the coils copper conductors is presently ongoing inside the vessel. The divertor coils power supplies have been produced in China in collaboration with the South-Western Institute of Physics, for delivery in Cadarache in spring 2016. 3 new CW ELM-resilient ICRH antennas are being manufactured in China in collaboration with the Institute of Plasma Physics of the Chinese Academy of Sciences, with a trial assembly of the first antenna planned in China in spring 2016. The procurement of the ITER-like divertor plasma facing units (PFUs), using the ITER tungsten monoblock technology, is underway in collaboration with the European and Japanese Domestic Agencies in charge of providing ITER divertor targets. Prototypes from ITER potential suppliers are in preparation and planned to be tested in the WEST tokamak environment. Series production will be launched afterwards and will not be available before 2018. For the other high heat flux plasma facing components, tungsten-coated technologies have been qualified on various substrates (CuCrZr, CFC, and Fine Grain Graphite). In particular, inertial graphite PFUs with tungsten coating from the Romanian National Institute for Laser, Plasma and Radiation Physics have been manufactured in order to complement the ITER-like PFU on the WEST lower divertor for the first phase of operation. Roughly 1/3 of the upper and lower divertor tungsten coated PFU have been received in Cadarache. The overall diagnostic layout has been finalized, with key diagnostics being implemented for monitoring of the divertor heat loads, plasma edge conditions and tungsten sources and transport (Langmuir probes, infrared PFC monitoring, calorimetry, visible spectroscopy ...). The data acquisition systems, the control, data access and communications (CODAC) is also being upgraded in partnership with the Institute for Plasma Research in India. A new plasma control system prototyping ITER requirements is being developed in collaboration with the Max Planck Institute for Plasma physics in Germany. First plasma is targeted for fall 2016. WEST provides relevant plasma conditions for validating PFU technology and exploring high heat flux operation and high particle fluence plasma wall interactions with tungsten. WEST operation is phased, to make the best use of the WEST ITER like PFU as they become available. In phase 1, WEST lower divertor will be operated with a mix of actively cooled ITER like PFU and inertial tungsten coated divertor start up elements. In this phase, plasma operation will be limited in time by the inertial divertor elements (typically ~5-10 s at high power). In phase 2, the full actively cooled ITER like divertor will be implemented, allowing long pulse operation up to 1000 s. High priority research areas of the PFU testing program will be described (focus on power handling performance of the ITER PFU during phase 1, high particle fluence interactions during phase 2). The WEST platform is open to ITER partners, and an international call for proposals has been send in January 2016. The paper will present an overview of the experimental program as derived from the call process.
5.3

Design and definition of a Divertor Tokamak Test facility

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One of the main challenges in the European roadmap toward the realisation of fusion energy with a demonstration plant DEMO [1] is to develop a heat and power exhaust system able to withstand the large loads expected in the divertor. In parallel with the programme to optimise the operation with a conventional divertor based on detached conditions to be tested on ITER, efforts are being devoted to the definition and the design of a “Divertor Tokamak Test facility” (DTT). Aim of DTT is to assess the set of possible alternative solutions including advanced magnetic configurations and liquid metal divertors. DTT should operate integrating the most relevant physics and technology issues, with significant power loads, flexible divertors, plasma edge and bulk conditions approaching as much as possible those planned for DEMO, at least in terms of dimensionless quantities. The machine parameters are selected so as to have a balance between these requirements and the need to realize the new experiment accomplishing the DEMO timescale within a reasonable budget: major radius $R=2.15$ m, aspect ratio $R/a = 3.1$, toroidal field $B_T=6$ T, plasma current $I_p=6$ MA, additional power $P_{Tot}=45$ MW. The machine will have the possibility to test several different magnetic divertor topologies (including single null, double null, snowflake, X-divertor) in reactor relevant regimes. Different plasma facing materials will be tested (tungsten, liquid metals) up to a power flow of the order of 20 MW/m². The main target of this experiment is the realization of an integrated solution (bulk and edge plasma) for the power exhaust in view of DEMO. According to the European roadmap, the DTT experiment should start its operation in 2022. To be coherent with this plan, the realization of the device will cover a time of around 7 years, starting from the first tender (during 2016) up to full commissioning and the first plasma (during 2022). The operations should then cover a period of more than 20 years, up to the initial phases of the DEMO realization. This talk presents the DTT proposal worked out by an International European Team of experts [2]. It demonstrates the possibility to set up a facility able to bridge the power handling gaps between the present day devices, ITER and DEMO within the European fusion development roadmap, which plays a crucial role for the development of one of the most promising technologies for an alternative, safe and sustainable energy source. [1] F. Romanelli et al., Fusion Electricity - A roadmap to the realisation of fusion energy, EFDA, November 2012, ISBN 978-3-00-040720-8 [2] A. Pizzuto (Ed.), DTT - Divertor Tokamak Test facility - Project Proposal, ENEA, July 2015, ISBN: 978-88-8286-318-0, http://fsnfusphy.frascati.enea.it/DTT/downloads/Report/DTT_ProjectProposal_July2015.pdf
The advanced stellarator concept beyond W7-X: motivation and options for a burning plasma stellarator

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One of the high-level missions of the European Roadmap to the realisation of fusion energy is to bring the HELIAS stellarator line to maturity. The near-term focus is the scientific exploitation of the Wendelstein 7-X experiment in order to assess stellarator optimization in view of economic operation of a stellarator fusion power plant. W7-X will play a decisive role for these studies but may be too small to explore all issues related with a burning-plasma in 3D geometry. Therefore, an intermediate-step burning-plasma stellarator appears prudent to mitigate the risks which would otherwise arise from the incomplete physics basis. A decision on the necessity of a burning-plasma experiment, however, must await the results of high-performance steady-state operation of W7-X and the fusion phase of ITER. However, in preparation of this review point and as a starting point for a more in-depth discussion of a research strategy, gaps in physics and engineering need to be investigated. Open aspects are the confinement of fast fusion-born particles and the behaviour of a burning plasma with considerable production of fusion power in a 3D magnetic topology. Apart from direct losses of fast particles, a sufficiently large pressure of fast particles may excite and interact with Alfvénic instabilities causing additional transport. As the confinement of the fast alpha particles is a key requirement for the self-sustained burn of a fusion power plant, a burning-plasma stellarator experiment allows clarifying these aspects for 3D magnetic configurations. From a systemic point of view, such a device also allows to elucidate on other uncertainties, e.g. the role of turbulent transport. It has recently been found that in W7-X the ITG mode is mostly located in a thin band on the outboard side of the torus, i.e. the properties of the magnetic configuration seem to play an important role, which can be incorporated in future optimization procedures. Investigation in a stellarator burning-plasma experiment allows, therefore, to obtain a clear systemic physics and engineering basis. Several different strategies could be followed for such a device ranging from a fast-track, cost-efficient device without blanket to a nearly DEMO-like machine requiring a full set of reactor systems. For each concept, a design analysis has been carried out using a systems code approach to define possible scenarios. The individual design points are compared in this common framework showing a factor of two differences in costs between the smallest reasonable (R=14m) and a DEMO-like (R=18m) device requiring in addition considerable technological development. It is expected that experience from the tokamak development (e.g. ITER) can be used to reduce the total effort. To substantiate these studies, further criteria are discussed to make a sensible choice on which design shall be followed. Especially the difference in technological readiness must be taken into account and it should be assessed to what degree synergy effects with the development towards a tokamak-DEMO can be expected.
Within the framework of the EUROfusion programme, a work-package of technology projects (WPJET3) is being carried out in conjunction with the planned DT experiment at JET with the objective of maximising the scientific and technological return of DT operations in support of ITER. To this purpose, experiments, analyses and studies are performed in the areas of neutronics, neutron induced activation and damage in ITER materials, nuclear safety, tritium retention and outgassing in plasma facing materials, and waste production and characterization. This overview presents the results achieved since the project start in preparation of DT operations. Accurate calibration of JET neutron detectors at 14-MeV neutron energy is needed to measure the fusion power and plasma ion parameters during DTE2, and fully exploit the available neutron budget thus obtaining a full scientific return for the investment in DTE2. The 14-MeV neutron calibration of JET has been designed using a suitable 14-MeV neutron generator, to be deployed by the JET remote handling system, and fulfilling all challenging requirements imposed by physics, safety and remote handling. The portable neutron generator purchased for this purpose has been calibrated and fully characterized to the required accuracy at a standard neutron facility using different measuring techniques. The JET calibration strategy has also been developed to benchmark the calibration procedure envisaged in ITER where neutron detectors have to provide, with accuracy better than 10%, not only the fusion power but also the amount of tritium burnt for tritium accountancy. Neutronics benchmark experiments are carried out during and after every experimental campaign at JET, and will continue in DTE2, with the objective of validating the neutronics codes and tools used in ITER nuclear analyses, thus reducing the related uncertainties and the associated risks in the machine operation and maintenance. In the Neutron Streaming experiment, new streaming paths have been recently investigated by measuring the neutron fluence along and outside diagnostics channels in the JET biological shield. In the Shutdown Dose Rate experiment, the gamma dose rate has been measured during non-operational periods at ex-vessel positions. In both cases, results are compared with calculations performed with the codes used in ITER analyses and, in general, the comparisons show a satisfactory agreement within the experimental uncertainties. Two laboratory scaled facilities have been designed and are currently being constructed to study the tritium retention and outgassing in samples of ITER plasma facing materials (Be, W) under controlled conditions: the short term outgassing and the retention under plasma relevant conditions will be studied in the Tritium Loading Facility, using an ion sputter gun, whereas those resulting from torus relevant conditions, including torus venting, will be investigated in the Tritium Soaking Facility. In this overview, the progress in the development of other technology projects is presented as well. They include the development of neutron detectors and methods for ITER TBM to be tested in JET, the collection of data on the occupational exposure, the preparation of measurements of neutron induced activity in ITER in-vessel materials, for the validation of numerical predictions of activation and dose rates in ITER, and of radiation damage in ITER functional materials up to 10-5 dpa during DTE2 with neutrons with a real fusion energy spectrum.
Tritium modeling in HCPB breeder blanket at a system level: a preliminary parametrical approach

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Tritium behavior in a breeding blanket is a key design issue because of its impact on safety and fuel-cycle best performance. Nowadays there are only few references and any fully validated tool with predictive capabilities. Considering the difficulty in handling tritium and its fundamental role inside a fusion reactor, it is intended to prepare a simulation tool for tritium transport. In this work a preliminary model for tritium transport at system level has been developed for the HCPB breeder concept (2014), focusing on the multi-physics of the release, diffusion, permeation, recombination phenomena. The numerical technique presented here is based on EcoSimPro simulation tool, a program with an object-oriented nature which offers the possibility of mixing various disciplines by robust equation-solving algorithms. The model is based on the integration of gas flow, concentration and tritium mass transport phenomena together with the isotope interactions between tritium atoms and the atoms of the purge gas flow. Some simplified assumptions have been adopted and the achieved results have been compared with others studies obtained by other programs. With this simplification in mind, this preliminary model includes a Tritium Extraction System and the counter of all the breeder modules present in a DEMO design blanket. The study is presented with several parametrical studies which aim is to identify the most relevant parameters for implementing breeder performances concerning tritium extraction and permeation. Significant variation in the permeation rate, as in the inventory of tritium in the different components, has been obtained especially from the variation of purge gas properties (pressure, flow rate, chemical composition). After this analysis, a quite low permeation factor per day has been found (about the 0.4\% of the diary T generation), confirming that HCPB model seem to be a quite robust breeder concept in terms of tritium radiological risk.
Progress of R&D on advanced tritium breeders for BA activity in EU and Japan

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Any demonstration power reactor (DEMO), which applies solid breeder blankets, requires “advanced tritium breeders” with high tritium breeding ratios and increased stability at high temperatures. However, the fabrication techniques of advanced tritium breeder pebbles have yet to be established. Therefore, the R&D on the fabrication technologies of the advanced tritium breeders and the characterization of the developed materials have been addressed in a collaborative effort between the EU and Japan (JA) in the DEMO R&D of the International Fusion Energy Research Centre (IFERC) project as a part of the Broader Approach (BA) activities from 2007 to 2016. In the EU, two-phase materials consisting of Li\(_4\)SiO\(_4\) and Li\(_2\)TiO\(_3\) were developed and pebbles are produced by a novel melt-based processing. By material as well as process optimizations, the product quality as well as the yield of the process could be improved. In the JA, Li\(_{2+x}\)TiO\(_{3+y}\) with Li\(_2\)ZrO\(_3\) (LTZO) was developed as new ceramic composites with both high stability and high Li density. The pebbles are fabricated by an emulsion method. With respect to the tritium release characteristics of the blanket, a grain size of less than 5 \(\mu\)m after sintering is anticipated. Therefore, the fabrication conditions were surveyed to fabricate such pebbles. Moreover, a number of collaborative works were performed. For instance, long-term annealing experiments of EU and JA pebbles were carried out, and the effects on the pebble quality were investigated. Also, the novel two-phase Li\(_4\)SiO\(_4\) pebbles with Li\(_2\)TiO\(_3\), Li\(_2\)AlO\(_2\) or Li\(_x\)La\(_y\)TiO\(_3\) were fabricated by the emulsion method as promising future breeder pebbles. This BA activity will be completed within this year, thus, we summarize the progress of R&D on advanced tritium breeders.
Design and R&D Activities of Fusion Breeder Blankets in China

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China has long been active in pushing forward the fusion energy development to the demonstration of electricity generation. Two experts' meetings were organized in 2014 by Ministry of Science and Technology (MOST) to seriously discuss the China's fusion roadmap in particular the design and construction of magnetic confinement fusion reactor beyond ITER. As one of the most challenging components in the fusion reactor, great efforts have been put on the development of breeder blanket which is the central part of fusion nuclear science and technology (FNST). Three blanket concepts have been mainly developed in China for China Fusion Engineering Test Reactor and DEMO, including Dual Functional Lead Lithium (DFLL), Helium Cooled Ceramic Breeder (HCCB), and Water Cooled Ceramic Breeder (WCCB). Moreover, there are also some other options in the early stage of concept design. In this paper, the blanket concept studies in China will be summarized, and the corresponding research and development activities will also be presented. The latest progress and technical challenges were emphasized in the fusion nuclear material development, the breeder and coolant technology and relevant test platforms, the Tritium technology achievement, and fusion nuclear safeguard, which will form the very basis of FNST for fusion blanket. Finally, the possible blanket development roadmap to DEMO in China will also be included as well as the international collaboration strategy. Keywords: Fusion Blanket, TBM, Development Roadmap
WCLL breeding blanket design and integration for DEMO 2015: status and perspectives

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Water-Cooled Lithium-Lead Breeding Blanket (WCLL) is considered a candidate option for European DEMO reactor. Starting from previous experiences in the frame of Power Plant Conceptual Studies within EUROfusion Consortium, ENEA and its linked third parties have proposed and are developing a multi-module blanket segment concept based on DEMO 2015 specifications. The layout of the module is based on horizontal (i.e. radial-toroidal) water cooling tubes in the Breeding Zone (BZ), and on Lithium Lead (PbLi) flowing in radial-poloidal direction. This design choice is driven by the rationale to have a modular design, where a basic geometry is repeated along the poloidal direction of the segment. The modules are connected with a back supporting structure, designed to withstand thermal and mechanical loads due to normal operation and selected postulated accidents. Water and PbLi manifolds are designed and integrated with a consistent primary heat transport system (PHTS), based on a reliable pressurized water reactor (PWR) operating experience, and the PbLi system. The pulsed operation of DEMO is addressed thanks to the adoption of a molten salt system. Rationale and features of current status of WCLL BB design are discussed and supported by thermo-mechanics, thermo-hydraulics and neutronics analyses. Open issues and areas of R&D needs are finally pointed out.
O1B.1

Uncertainties in power plant design point evaluations

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When designing a new large experimental device, extrapolation from current knowledge and rules into unexplored design space is unavoidable, and predicting the behaviour of a new device is therefore subject to significant uncertainties. This makes it difficult to determine an optimal design. For conceptual fusion power plants, a further concern is the large possible variation in expected plasma performance in fusion devices beyond the currently probed range. In particular, extrapolating scaling laws like for the L-H threshold or the confinement time hold large uncertainties as far as the predicted physics are concerned. On the technology side, predictions for the efficiency of the heating and current drive system or the allowable stresses on the magnet systems are still variable and can potentially influence the optimal machine design. In this work, we evaluate the effects of both physical and technological uncertainties in the current European pulsed DEMO designs (nominally 500MW net electrical power, 2 hour pulse length). We use a Monte-Carlo method in combination with the systems modelling code PROCESS to map out the probable machine performance. The results show the likelihood of current designs fulfilling the high-level goals of providing several hundred MW of net electricity while maintaining a reasonable pulse length, and we give recommendations of how to effectively minimise existing design uncertainties.
O1B.2

Long pulse and high performance discharges in KSTAR in preparing for ITER and beyond

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Extending high performance plasma discharge into long pulse steady-state operation is one of the urgent issues to be solved in preparing the ITER and fusion reactor. The KSTAR device is one of the best engineered superconducting tokamak devices which is good for exploring the science and technologies for the high performance steady-state operation due to lots of its unique features such as extremely well defined for 3D field research with low-level of intrinsic error field and versatile in-vessel control coil, and advanced image diagnostics. In the recent experiment in the KSTAR, the H-mode plasma discharge has been extended in pulse length up to 55s at 0.5 MA in plasma current and 2.9 T in toroidal field, which was the longest H-mode discharge in tokamaks so far. The fully non-inductive operation discharge were attempted to explore the steady-state operation mode. The first fully inductive operation was achieved at the reduced plasma current of 0.4 MA and the plasma performance was relatively high ($b_N \sim 2.1$ and $b_P \sim 3.0$). In the upcoming KSTAR 2016 campaign, we expect more improved plasma discharge up to 1 MA, and stabilized fully non-inductive scenario will be revisited. In this paper, the recent progress in high performance long pulse discharge will be introduced as well as the exploring the the scientific and engineering mechanism.
Design of machine upgrades for the RFX-mod experiment

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After 10 years of operation since its major modification, an upgrade of the RFX-mod experiment is presently under design. The main objectives are the improvement of the control of magnetic confinement, plasma density and plasma wall interaction in both RFP and Tokamak configuration. The main design driver requirement for the improvement of the magnetic confinement control is the enhancement of the 'plasma-shell proximity', to reduce the deformation of the last close magnetic surface. This requirement calls for the removal of the present vacuum vessel, the fastening of a new first wall to the existing copper stabilising shell, including some poloidal and toroidal gap modifications, and the modification of the present toroidal vacuum vessel to provide the function of vacuum barrier. The critical aspects of the new torus assembly are the development of composite ceramic/metal joint solutions for the equatorial and poloidal joints of the toroidal support structure, in order to guarantee the vacuum boundary and the penetration of electro-magnetic fields within the plasma chamber. Moreover the new configuration requires the integration of a further active control system of local magnetic field at the poloidal gaps, in addition to the existing set of saddle coils distributed around the whole torus. For the enhancement of the control of plasma density and plasma wall interaction two main improvements are proposed: a new first wall with higher thermal conductivity material and optimized shape, to withstand the severe power load experienced in RFX, and a new distributed glow discharge cleaning system, to improve uniformity and frequency of wall conditioning. Finally an upgrade of the plasma diagnostic system, both in-vessel and ex-vessel, is also included in the proposal. The paper will present an overview of the engineering design of the new components of the RFX-mod and their integration with the existing machine assembly.
Manufacturing of the JT60-SA cryostat vessel body cylindrical section

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The JT-60SA project implemented by Japan and Europe is progressing on schedule towards the first plasma in 2019. Spain (Ciemat) is in charge of the design and manufacturing of the cryostat. The JT-60SA cryostat is a stainless steel vacuum vessel (14m diameter, 16m height) which encloses the tokamak providing the vacuum environment ($10^{-3}$-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$^{[0+0.05]}$ wt%) with a permeability ($\mu_{rel}$) below 1.1. The CB was manufactured and assembled in situ in 2013, while the CVBC is currently under manufacturing by a Spanish company and it is expected to be delivered in Naka next year 2017. The CVBCS is made of a single wall stainless steel shell with a thickness of 34mm. This paper summarizes the manufacturing of the JT-60SA CVBCS. The manufacturing includes the assembly and testing at the manufacturer workshop as well as the packaging of the component. Packaging must be suitable for shipping the component to a port of entry in Japan. The reference code being used for the manufacturing is ASME code 2007.
Characterization of the ITER CS conductor and projection to the ITER CS performance

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The ITER Central Solenoid (CS) is one of the critical elements of the machine. The CS conductor went through an intense optimization and qualification program, which included characterization of the strands, a conductor straight short sample testing in the SULTAN facility at the Swiss Plasma Center (SPC), Villigen, Switzerland, and a single-layer CS Insert coil recently tested in the Central Solenoid Model Coil (CSMC) facility in Naka, Japan. We obtained valuable data in a wide range of the parameters (current, magnetic field, temperature, and strain), which allowed a credible characterization of the CS conductor in different conditions. Using this characterization, we will make a projection to the performance of the CS in the ITER reference scenario. This manuscript has been authored by UT-Battelle, LLC under Contract No. DE-AC05-00OR22725 and by Lawrence Livermore National Security, LLC, under Contract No. DE-AC52-07NA27344 with the U.S. Department of Energy. The United States Government retains and the publisher, by accepting the article for publication, acknowledges that the United States Government retains a non-exclusive, paid-up, irrevocable, world-wide license to publish or reproduce the published form of this manuscript, or allow others to do so, for United States Government purposes. The Department of Energy will provide public access to these results of federally sponsored research in accordance with the DOE Public Access Plan(http://energy.gov/downloads/doe-public-access-plan).
O1C.2

Initial Operation of 3 MW Dual Output High Voltage Power Supply with IC RF System

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Pulse Step Modulation (PSM) based High Voltage Power Supply (HVPS) are widely used in applications viz. Broadcast transmitters, Particle accelerators and Neutral Beam Injectors because of inherent advantages of modular structure, high accuracy and efficiency, low ripple and fast dynamics. Typical IC RF system composed of cascaded connection of Driver stage (70 kW RF output) and End stage (1500 kW RF Output) would need two power supplies. A novel concept of tapping two outputs from single PSM based HVPS is attempted for the first time. PSM based 3MW HVPS is developed with dual output to feed anode voltage of Driver (up to 18kV) and Final Stage (up to 27kV). Controller for the HVPS is designed to suit requirements during local operation and remote operation. In Local operation mode, protection interlocks with in the HVPS are integrated while remote mode has additional interlocks from IC RF Amplifiers system. Field integration of the Controller is achieved through Fiber optical links to avoid EM interferences. Present article discusses initial experience of dual output HVPS with IC RF amplifiers, pre-integration validation campaigns including qualification for energy limits. The paper also presents the performance of HVPS at full rated capacity of 2800kW continuous duty to support 1.5 MW IC RF system.
O1C.3

Results of the first JT-60 SA TF coils tests in the Cold Test Facility

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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 are provided by European industry and tested in a Cold Test Facility (CTF) at CEA Saclay. The first coils were 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 first 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. This paper will give an overview of the main experimental results obtained during these tests. These results will be analyzed and discussed in the light of the expected TF coils performances.
O1C.4

Effect of cycling load on performance of strands made by twisted-stacks of HTS tapes

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Various tests performed with full-size 60 kA HTS cable prototypes for fusion magnets in EDIPO test facility demonstrated that design of HTS strand proposed at Swiss Plasma Center – stack of HTS tapes twisted and soldered between two copper profiles – is applicable for high-current fusion cables, but additional mechanical reinforcement is still needed. Based on experimentally obtained correlation between the performances of cable prototypes at different operating conditions, further key investigation of cycling transverse load on the strand performance was performed at 77 K. Aiming to obtain a strand design able to withstand a continuous cycling load operation of some thousand cycles, following aspects of the strand design were studied – geometry of the strand, preliminary annealing of the copper profiles, manufacture of HTS tapes. Appropriate FEM mechanical modelling is performed and reported, showing a good agreement with the test results. Based on the obtained data, next design of HTS cable prototype will be discussed.
Nanoscale characterisation of radiation damage in tungsten alloys

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Tungsten is the leading candidate material for plasma facing applications in future tokamak systems, due to its high melting point, good sputtering resistance and low activity after irradiation. Despite this there has been a significant lack of study of the effect of transmutation products on the post irradiation mechanical behaviour of tungsten-based alloy systems. This will be key to understanding component lifetimes in future devices. This study examines the formation of solute clusters and the associated hardening in W-2 at.%Re, W-2 at.%Ta, W-1 at.%Re-1 at.%Os and W-1 at.%Re-1 at.%Ta alloys induced by 2 MeV W+ ion irradiation at 573 and 773 K to damage levels of 33 dpa. Such clusters are known precursors to the formation of embrittling precipitates, which are likely to be the life-limiting factor in the operation of fusion reactor components. Due to the shallow depth of the damage layers, atom probe tomography was used to study chemical segregation and nanoindentation was used to measure increases in hardness due to irradiation. The presence of osmium significantly increased post-irradiation hardening compared to rhenium- and tantalum-containing binary alloys (a peak hardness of 12 GPa, compared to 9 GPa for the binary alloys). Atom probe tomography analysis revealed solute clustering in rhenium- and osmium-containing alloys, with the size and number densities strongly dependent on alloy composition and irradiation temperature. The highest cluster number density was found in the ternary tungsten-rhenium-osmium alloy irradiated at 773 K. In this ternary alloy, osmium was found to cluster preferentially compared to rhenium. No clustering of tantalum atoms was seen in binary or ternary alloys. The implications of these results for the structural integrity of fusion reactor components will be discussed, and future research questions regarding phase stability in these systems will be identified.
O2A.2

Effect of Aging on Microstructural Evolution and Mechanical Properties in Ti-Bearing RAFM steels

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Microstructural evolution and mechanical properties of Ti-bearing RAFM steels were investigated after aging at 550 °C for 0 ~ 1000 hr. All samples with Ti were prepared using vacuum induction melting furnace and hot rolling process, followed by heat treatment in normalizing and tempering. Microstructures including precipitates, fractured surfaces and cross-sectional microstructures were observed using a scanning electron microscopy (SEM) and a transmission electron microscopy (TEM). Mechanical properties were measured by charpy impact test as well as tensile test. It was experimentally indicated that mechanical properties of Ti-added RAFM steels after aging showed degradation less than those of conventional RAFM steels due to fine (Ti,W)C carbides with higher thermal stability.
O2A.3

Grain size influence on the response of SPS tungsten to ELM-like thermal shock loading

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Tungsten is the main candidate material for the plasma facing components of future fusion devices. During operation, these components will be subject to severe conditions, involving both steady state and transient heat loads as well as high particle fluxes. These may lead to surface and structure modifications which influence their performance and lifetime. Therefore, it is necessary to study these effects to understand the physical processes and predict the behavior in such extreme environments. At the same time, various novel materials are being developed with the aim of improved properties and lifetime. Among the processing techniques, spark plasma sintering (SPS) is promising thanks to the relatively low temperatures and shorter sintering times compared to traditional powder metallurgy techniques. In this work the influence of tungsten microstructure on its response to combined deuterium plasma and laser loading were studied. Set of tungsten samples with variable grain size was prepared by SPS through variation of the fabrication parameters. These were exposed to steady state deuterium plasma beam and high energy heat pulses (100 laser pulses of 1 ms duration and power density of 0.76 GW/m\textsuperscript{2}) in the PSI-2 device, simulating tokamak operation in the ELMy H-mode. To discern the contribution of these two exposure modes, both sequential and simultaneous loading was performed. A comprehensive post-mortem characterization of the exposed samples was carried out. Due to the exposure, sample surfaces were roughened, as-prepared grains were recovered; in few isolated cases, cracks were formed. Post-irradiation analysis revealed activation of in-grain slip systems within the loaded surfaces. Damage features were found to depend on more fabrication parameters than grain size. For example, cracks apparently initiated on grinding grooves on (purposefully) unpolished surfaces. Depending on the microstructure, the performance of the SPS tungsten was comparable or better than that of a reference tungsten material (“ITER grade”).
Advanced Materials for a Damage Resilient Divertor Concept for DEMO

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Material issues pose significant challenges for future fusion reactors like DEMO. When using materials in a fusion environment a highly integrated approach is required. Cracking, oxidation and fuel management are driving issues when deciding for new materials. Neutron induced effects e.g. transmutation adding to embrittlement are crucial to material performance. Here advanced materials e.g. Wf/W or Cu/W composites allow the step towards a fusion reactor. Recent developments in the area of multi-fibre powder-metallurgical Wf/W will be presented showing a possible path towards a component based on standard tungsten production technologies. Spark-plasma sintering is used as production route to achieve fully-dense materials. Initial mechanical tests and microstructural analyses show potential for pseudo-ductile behavior of materials with a reasonable (30%) fibre fraction. In the as-fabricated condition samples showed step-wise cracking while the material is still able to bear rising load, the typical pseudo ductile behavior of a composite. The optimization of the interfaces is a crucial aspect when establishing this behavior. As damage resilient materials, with an increased operational temperature range facilitate component design with higher exhaust capabilities we propose to utilize the Wf/W composite approach together with self-passivating alloying concepts to maximize the potential of W-based-PFCs. The lifetime influenced by erosion, creep, thermal fatigue, and embrittlement, needs to be compatible to the requirements from steady state operation. The maximization of operational performance can only be achieved, if improvements of material properties, mechanical and thermal, are well balanced and do not occur at the expense of each other. Wf/W contributes here to advanced material strength and crack resilience even after embrittlement. Together with W/Cu composites at the coolant level high-performance components can be developed. Rigorous testing with respect to PWI and high heat-flux performance are planned to have prototype components available within 5 years for application in existing fusion devices.
The conceptual design of the European DEMO power reactor is under development as part of the EUROfusion Consortium. DEMO is a high fusion power, long-pulsed, tritium self-sufficient device, and hence amongst the most critical and high-risk technologies are the divertor and main chamber plasma-facing components (PFCs). These PFCs must operate reliably under an extreme surface heat and particle flux while surviving intense neutron radiation, and must also allow sufficient high energy neutron transmission to the tritium breeding blankets. In addition, a preliminary assessment of wall surface loads (and their uncertainty) has led to the anticipated requirement for high heat flux PFCs in certain regions of the main chamber wall, perhaps embodied as discretely placed limiters. Such challenging requirements and conditions have necessitated wide engineering design exploration studies, and these have started to yield promising PFC designs. In this paper, we present engineering concepts of divertor and first wall PFCs which, compared to baseline designs, are intended to improve power handling or extend operational life. Design by analysis is used routinely with the objective of reducing stress in the structure, the tungsten armour, and/or the interface between these materials. A number of designs are featured here. An update is given on the Thermal Break divertor PFC design, including results of fabrication trials and high heat flux mock-up manufacturing. A discrete limiter PFC is outlined, which makes use of the thermal inertia of tungsten to improve power handling for short durations. Progress is reported with the de-coupled first wall ‘finger’ and monoblock PFC designs. The potential and limits of each design are reviewed and the outlook for future work is described.
O2B.2

Development and analyses of self-passivating tungsten alloys for DEMO operational and accidental conditions

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Tungsten is considered the main candidate material for the first-wall in DEMO for its high melting point, low erosion yield and low fuel retention. Nevertheless, it can cause a substantial safety issue in a loss-of-coolant accident (LOCA) in combination with air ingress into the plasma vessel, due to formation and evaporation of volatile neutron activated tungsten oxide. Self-passivating tungsten alloys introduce a passive safety mechanism by forming a stable chromic oxide layer on the surface acting as a diffusion barrier for oxygen and preventing the formation of tungsten oxide. In this contribution self-passivating tungsten alloys optimised for oxidation resistance containing 12wt.% chromium and ˜0.4wt.% yttrium are investigated under conditions of argon-oxygen, argon-water and nitrogen-oxygen-water atmospheres at different partial pressures and temperatures ranging from 1073 to 1473K. Thin films with 3.5µm thickness produced by magnetron sputtering are used as a model system. The oxidation resistance of these films in an argon-20vol.% oxygen atmosphere is sufficient to prevent formation and release of tungsten oxide for more than 60h at 1073K, and for 2.5h at 1273K to up to 9h by doubling the film thickness. All following the favoured parabolic oxidation regime. Assuming an armour thickness of 2mm, mitigation of tungsten oxide release for several years under the conditions of a LOCA with air ingress is predicted. In argon-water the alloy shows linear oxidation behaviour without release or formation of tungsten oxide within 2h at 1273K. The evaporation of chromium in nitrogen-oxygen-water atmosphere at ≤1273K will be discussed. A deeper understanding of the governing processes for oxygen/chromium diffusion under different atmospheres will be shown, supported by SEM/TEM/EDX, XRD, TGA and SIMS measurements. Furthermore, the production of W-Cr-Y bulk-samples is ongoing using mechanical alloying. The plasma performance and thermo-mechanical characteristics of these bulk-samples will be presented.
O2B.3

**Overview of the different processes of tungsten coating implemented into WEST tokamak**

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The main objective of the WEST (W Environment in Steady-state Tokamak) project is to fabricate and test an ITER-like actively cooled tungsten divertor to mitigate the risks for ITER. Concerning the others Plasma Facing Components (PFC), they will also be modified and coated with W to transform Tore Supra into a fully metallic environment. Solutions had been developed with three different suppliers, taking into account the specifications of each kind of PFC: heat flux up to 10MW/m², complex geometries including components 1m long and different substrates, CuCrZr for actively cooled PFC, graphite and CFC for other components. Plasma Vapour Deposition (PVD) is used on the main PFCs: lower divertor and bumpers (W/Mo on graphite or CFC substrate, as developed for ASDEX Upgrade), upper divertor, baffle and Vertical Displacement Event protection (W on CuCrZr substrate, new development). Vapour Plasma Deposition (VPS) technique is used on the antenna protection limiter due to their large size (W/Mo on CFC substrate, new development). For each type of coating, specifications are achieved in terms of density (>90%), homogeneity, impurity content (lower than a few percent) and absence of cracks and scratches. Several high heat flux tests have been performed on the different coating and substrates to validate each process. Consequently limits in terms of surface temperature have been defined, especially for the coating on graphite and CFC. Indeed, in this case limitations primarily come from the coating itself (around 1200°C) while for the actively cooled components it comes from the CuCrZr substrate (around 450°C). The paper gives an overview on all these different processes and associated validation programme and concludes on the adequacy of the W coating used in WEST tokamak.
Development of the flowing liquid lithium limiter for EAST tokamak

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Lithium coating technology and flowing liquid lithium limiter (Flili) have been applied on HT-7 tokamak and many significant results have been obtained. A Flili for exploring lithium as potential plasma facing material was designed and manufactured for EAST tokamak, it is applied on the concept of the thin flowing film which had been successfully tested in HT-7 tokamak. The Flili of EAST mainly composed of distributor, collector, guide plate, heater, cooling system and one in-vessel electro-magnetic pump installed on the bottom of limiter which is totally different from it applied on the Flili of HT-7. The in-vessel electro-magnetic pump can make liquid lithium circulate from the bottom collector to the distributor of the limiter, which will make Flili steady-state operation and also drastically reduce the amount of lithium used for experiment. The Flili can be moved along the guide rail into vacuum vessel to meet various plasma scenarios by driver system. A Flili has been successfully tested in EAST tokamak in 2015, significant achievements were achieved including the in-vessel electro-magnetic pump, working with the toroidal magnetic field of the EAST device, quite reliable to control the lithium circulation flow and flow speed, and circulating lithium layer with a thickness of lower 0.1mm and a flow rate 2cm\textsuperscript{3}/s as well. Besides those, some problems also were exposed, which requires improvement of Flili for next campaign in EAST. The paper is mainly introduce covering the design, manufacturing and improvements of the Flili. The improvement Flili is foreseen to be installed in EAST in 2016, and participated in operation for test.
Manufacturing and test of the first ELM resilient long pulse ICRH antenna for WEST

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One of key missions of WEST (Tungsten (W) Environment in Steady-state Tokamak) is to pave the way towards the ITER actively cooled tungsten divertor procurement and operation. WEST PFC will operate in ITER conditions, i.e. with a heat flux on the divertor target of 10MW/m² during 1000s and 20MW/m² during a few tens of seconds. To achieve such heat flux levels, both Lower Hybrid Current Drive (LHCD) and Ion Cyclotron Resonant Heating (ICRH) systems are used. ICRH system is designed to operate at a power level of 9MW during 30s or 3MW during 1000s using three antennas. The WEST ICRH antennas [1-4] have been designed in a European collaboration and are now under fabrication at CAS/ASIPP, Hefei, within the framework of the Associated Laboratory IRFM-ASIPP. Since the WEST ICRH antennas have a CW operation requirement at high power, stringent specifications for the manufacturing process are necessary, such as material choice, high precision machining processes, etc. A specific welding sequence study was carried out for reducing the deformation during welding, dedicated fixation tools and jigs were designed for the fabrication and assembly, and the assembly workflow was optimized. This paper gives an overview of the manufacturing of the ICRH antenna sub-assemblies and the associated intermediate tests carried out. The work methods implemented for the pre-assembly are also presented. [1] Z. Chen et al., Fusion Eng. Des. 94 (2015) 82 [2] W. Helou et al., Fusion Eng. Des. 96-97 (2015) 473 [3] K. Vulliez et al., Fusion Eng. Des. 96-97 (2015) 611 [4] J. Hillairet et al., Proc. 21st Topical Conf. on Radiofrequency Power in Plasmas, Lake Arrowhead, CA (2015)
O2C.2

Development of long-pulse high-power-density negative ion beams with a multi-aperture multi-grid accelerator

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Acceleration of high-power-density negative ion beams of \(\sim 180 \text{ MW/m}^2\) have been achieved up to 60 s for the first time. Because the achieved power density was comparable to ITER accelerator, and accelerated energy density of 10800 MJ/m\(^2\) is much higher than that for JT-60SA of 6500 MJ/m\(^2\), this achievement is one of promising results to overcome common issues for the heating neutral beams on JT-60SA and ITER having same accelerator concept. Formerly, even after voltage holding capability and beam steering were improved by adjusting the gap length and compensating the beam deflections, the pulse length of over 100 MW/m\(^2\) beams was limited below 1 s due to breakdowns at acceleration grids. The breakdown was induced by impacts of negative ions and/or secondary electrons to the grids. In order to suppress this, one of remaining issues was the configuration of apertures on the grids, since the impact of the negative ions on the top surface of the grids had been minimized by the optimization of the beam steering. This time, the acceptance of the negative ions to the downstream side was improved. At first, the thickness of the grids has been reduced from 20mm to 10mm, and the shape of the apertures has been changed from straight to conical type. Moreover, the diameter of the apertures was enlarged from 14mm to 16mm for the downstream grids which accelerated high energy beam. After these modifications, because of the reduction of breakdowns with beam, acceleration of 1 MeV beam has been realized in shorter conditioning time of 4 days with cesium operation than the previous one of 10 days, and finally long-pulse acceleration of 60 s was obtained without any breakdown. So far, no degradation has been observed in terms of the negative ion acceleration during long pulse.
O2C.3

Development of a high power helicon system for DIII-D


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A new mechanism for driving current off-axis in high beta tokamaks using fast electromagnetic waves, called Helicons, will be experimentally tested for the first time in the DIII-D tokamak. This method is calculated to be more efficient than current drive using electron cyclotron waves or neutral beam injection, and it may be well suited to reactor-like configurations [1]. A low power (100 W) 476 MHz “combline” antenna, consisting of 12 inductively coupled, electrostatically shielded, modular resonators [2], was recently installed in DIII-D. Initial operation showed that the plasma operating conditions were achieved under which helicon waves can be launched. Plasma operations also showed that the location of the antenna has not reduced the performance of, or introduce excessive impurities into, the discharges produced in DIII-D. The development of a high power (1 MW) Helicon system is underway. This antenna consists of 35 modules mounted on the inside of the outer wall of the vacuum vessel slightly above the midplane. Carbon tiles around the antenna protect the antenna from thermal plasma streaming along field lines. A 1.2 MW, 476 MHz klystron system will be transferred from the Stanford Linear Accelerator to DIII-D to provide the RF input power to the antenna. A description of the design and fabrication of high power antenna and the RF feeds, the klystron and RF distribution systems, and their installation will be presented. This work is supported by the US DOE under DE-FC02-04ER546981 and DE-AC02-98CH10882. [1] J. Tooker, P. Huynh., J. Fusion Eng. Des., 88 (2013), 521 [2] J. Tooker, et al., Proceedings of the 26th IEEE Symposium on Fusion Engineering (SOFE), May 31 – June 4, 2015, Austin, TX
0-D Physical Design for the Heating and Current Drive System of CFETR

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As the next step for the fusion energy in China beyond ITER, the China Fusion Engineering Text Reactor (CFETR) aims to operate with duty time as 0.3–0.5, means that CFETR should operate at steady-state scenario. This provides a great challenge for the physical design of the heating the current driving system. In general, four different kinds of method as NBI, ECH, LHW and ICRH have been developed in worldwide for heating plasma and driving current. Considering the characteristics of each H&CD system, we provide two design solutions as the one with NBI and all-wave solution. For the solution with NBI, the total design power is 73MW with 33MW NBI, 20MW LHW and 20MW ECRH; For all-wave solution, the total design power is 80MW with 20MW LHW, 40MW ECRH and 20MW ICRH. Those two solutions can satisfy the heating and steady-state operating aims of the CFETR through the 0-D physical design.
Topology design of the in-vessel ITER plasma-position reflectometry antennas

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The ITER Plasma Position Reflectometry (PPR) system will be used to estimate the distance between the position of the magnetic separatrix and the first-wall at four pre-defined locations also known as gaps 3, 4, 5, and 6, complementing the information provided by the magnetic diagnostics. For gaps 4 and 6, the antennas are to be installed in-vessel between two blanket shield modules. The microwave signal is routed to/from the antennas using rectangular oversized waveguides that enter/exit the vacuum vessel through feed-outs located in upper ports 01 and 14, respectively. The antennas and adjacent waveguides are in direct sight to the plasma through cut-outs in the blanket shield modules and are subject to plasma radiation, neutronics loads, and stray-radiation from ECH that may cause excessive temperatures and stresses. Although the antenna assembly is designed in such way that it can be remotely installed and removed, it should withstand the maintenance period. Therefore, one of the initial constraints is the topology design of the antennas that arises from thermal-structural simulations. Here, we report on the topology optimization of the antennas that is conducted to find suitable material distributions in the antennas, while maintaining their internal geometry that is relevant for microwave diagnostics. The results obtained through the analysis presented here may be used as input in the global integrity analysis of the in-vessel components of the ITER PPR system.
Current status of the design of the ITER bolometer diagnostic

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The ITER bolometer diagnostic shall provide the measurement of the total radiation emitted from the plasma, a part of the overall energy balance. About 500 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 LOS are bundled in up to 100 individual cameras, which will be located behind blanket modules on the vacuum vessel wall, in five divertor cassettes, in two upper port plugs and in one equatorial port plug. For all major design issues solutions have been elaborated and will be presented: During a German nationally funded project generic designs for all major diagnostic components have been developed in close collaboration between IPP and its partners with the aim to enable reliable measurements under the harsh loads of the ITER environment. The design of collimators has been developed and tested on prototypes to provide an exact definition of the required viewing cones of about 1° while reliably reducing reflections, stray light and microwave stray radiation. A 3D-shaped ceramic printed circuit board is proposed to hold the sensor, orient it as desired, and provide good thermal contact as well as the bridge for electrically connecting external signal cables to the meanders on the sensor. The design of the camera housing for vacuum vessel and divertor cameras has been optimized for improved management of the thermal heat flow, supported by tests defining material properties and verifying analysis. Additionally, methods have been developed to derive the main design parameters of cameras and decide if pin-hole or collimator type is more advantageous. Recently, the system-level design phase within a framework partnership agreement with F4E started and uses the achieved results to define interfaces and designs for the specific locations in ITER.
Model identification of the temperature on the FTU liquid lithium limiter

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One of the main research lines currently investigated within the FTU programs is the possibility to adopt a technology based on liquid metals as first plasma wall. More particularly, the main attention has been devoted to the analysis of plasma performances when using a liquid lithium limiter (LLL) device. The control of the limiter surface temperature reveals to be a fundamental aspect of the LLL operativity, hence a model able to link the different physical observables to the thermal behavior of the LLL has to be identified. The temperature over the surface of the LLL in FTU is monitored through an infrared thermocamera located inside the vacuum chamber. We performed a complete analysis of the datasets collected at FTU during each experiment, evaluating the correlation coefficients with respect to the spatio-temporal distribution of the temperature over the limiter surface as measured by the infrared camera. The proposed approach relies on the training of an artificial neural network (ANN) with the aim of interpolating the dynamical relationship between the identified quantities and the thermal process with a nonlinear, autoregressive moving average model. The evaluation of these preliminary results, which are able to show an overall good estimation of the dynamical model regulating the thermal process over the limiter surface, suggests that the highly nonlinear dynamics regulating the interplay between plasma and thermal behavior of the limiter can be successfully modeled with an ANN.
System level design of the ITER Equatorial visible/infrared Wide Angle Viewing System

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The equatorial visible infrared wide angle viewing system (WAVS) is one of the key diagnostics in ITER aiming at the machine protection and plasma control. Those two main functions are achieved by means of infrared thermography and visible observation of the main plasma facing components. The diagnostic is composed of 15 lines of sight integrated in 4 equatorial port plugs allowing coverage of about 80% of the vacuum vessel. This diagnostic will be composed of 30 optical systems distributed under vacuum in the equatorial Port Plug and in air in the Interspace, in the Port Cell and in the diagnostic building. The data generated by the cameras will be processed and transferred among others to the Plasma Control System for real time control. Prior to the start of the design of such a complex system, a system level design (SLD) has been performed aiming at the identification of the requirements, the definition of the sub-systems and their functions, the location of the different components and the identification of the interfaces. The first phase of the SLD gathered the requirements. From the analysis of the project documentation, requirements applicable to the WAVS have been identified and implemented in a standard framework, allowing traceability of the requirements and their evolution during the project life. During a second phase, based on a functional analysis, translating requirements into functions from highest level (functional) down to components (technical), different architecture options have been produced. Compared to the initial architecture, alternatives are proposed in three areas: Port plug (substitution of the on-axis Cassegrain by an off-axis Gregorian telescope, a-focal optical interface), Interspace (suppression of refractive optics and Cassegrain telescopes) and Port Cell (separation of IR-Vis). The paper will present the details of the system level design approach for the WAVS project and the new architecture proposals.
O3B.1

The JET Materials Detritiation Facility for reducing radioactive waste liabilities

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Radioactive waste arisings from JET operations are projected to contain approximately 25t of non-incinerable Intermediate Level Waste (ILW) with tritium levels > 12 kBq/g. This originates primarily from plasma facing components, specifically the divertor (MKIIa) used during the JET Deuterium Tritium Experiment in 1997 (DTE1). As current UK regulations do not allow off-site disposal of ILW and restricts the time period waste may be stored, the only options available to the UKAEA for the MKIIa divertor is either off-site storage pending construction of the UK’s deep geological disposal facility (GDF), or, following a detritiation process on/off-site, to dispose as Low Level Waste (LLW - < 12 kBq/g). Following a sampling and analysis campaign on a number of tiles and carriers a process using a thermal bake at 1273K in air for 6 hours was tested indicating a detritiation efficiency between 94.08% and 99.97% for the carbon tiles, and between 99.990% and 99.998% for the inconel tile carriers. A cost analysis was completed for performing detritiation off-site compared to constructing and running an in-house detritiation facility. This indicated costs of £284/kg for disposal to the GDF and £304/kg and £179/kg for off and on-site detritiation respectively. A detritiation facility is planned to be constructed at JET and is due for completion in 2017 with an estimated throughput of at least 14,000 kg per year with the released tritium being collected and recycled via the JET Water detritiation system (WDS). The effective waste treatment will reduce the radioactive waste liability, lower costs, reduce the environmental impact and demonstrate best available technique (BAT). Detritiation of fusion waste on an industrial scale is applicable to ITER to reduce storage complexity and cost and is demonstrative of the process required by DEMO for a closed fuel cycle in a commercial fusion power station.
O3B.2

Options for a high heat flux enabled helium cooled first wall for DEMO

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Helium is considered as coolant in the plasma facing first wall of several blanket concepts for DEMO fusion reactors, due to the favorable properties of chemical inertness, no activation, comparatively low effort to remove tritium, no chemical corrosion and a flexible temperature range. Design analyses for the ITER Test Blanket Modules done by several design teams have shown ability to use helium cooled first walls with heat flux densities of 0.5MW/m$^2$. Investigations on the heat loads coming from the plasma are ongoing for current EU DEMO concepts. Typical steady state loads are predicted around 0.3MW/m$^2$, but peak values could reach and excess 1MW/m$^2$ near the lower and upper X-points, depending on the chosen first wall shape, magnetic configuration and assumptions on power fall off lengths in the scrape off layer of the plasma. Even higher short-term transient loads can be expected. Several modifications to the helium cooled first wall channel shape were investigated in terms of heat transfer and pressure drop by computational fluid dynamics and experiments. The results indicate an excellent performance of transversal ribs (wall mounted or detached) and other mixing devices in the first wall cooling channels, enabling augmented heat flux capabilities with tolerable pumping power increases. Additional to decreasing the structural material peak temperature, a fine tuned application of heat transfer enhanced surfaces can also reduce the temperature spread within the component and thus reduce the thermo-mechanical stresses. The applicability of the investigated channel surfaces is closely linked to the manufacturing strategy of the first wall. Several manufacturing methods are explored, enabling the application of the suggested heat transfer enhanced first wall channels.
SiC-based sandwich material for Flow Channel Inserts in DCLL blankets: manufacturing, characterization, corrosion tests

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Flow Channel Inserts (FCI) are key elements in a Dual Coolant Lead Lithium blanket concept for DEMO, since they provide the required thermal and electrical insulation between the He cooled structural steel and the hot liquid Pb-15.7Li flowing at around 700°C, and minimize MHD pressure loss. FCIs must be inert in contact with Pb-15.7Li and show low tritium permeability. In addition, FCIs have to exhibit sufficient mechanical strength to withstand thermal gradients during operation. SiC fulfils the operational requirements for FCIs. Besides, porous SiC is an attractive candidate to obtain a thermally- and electrically- low conducting structure. To prevent tritium permeation and corrosion by Pb-15.7Li a dense SiC coating shall be applied on the porous SiC. In this work a SiC-based sandwich material consisting of a porous SiC core covered by a dense CVD-SiC layer is proposed. The production method of the porous SiC consists in combining the particle size of the starting mixture of SiC powder and a carbonaceous sacrificial phase (which is removed after sintering by oxidation), in such a way that a honeycomb microstructure –mechanically more resistant- is achieved. The porosity of this tailored microstructure results in low enough thermal conductivity for sufficient thermal isolation through a core thickness of 5 mm, as determined by thermo-mechanical analysis. This analysis provides also the optimum thickness of the dense CVD-SiC layer for minimizing thermal stresses. In this paper a study of the microstructure, thermal and electrical conductivities and flexural strength of the sandwich material with a dense CVD-SiC coating of 200 µm is presented. In order to ascertain whether this dense layer thickness is enough to withstand the contact with the hot PbLi without corrosion damage, laboratory tests have been performed in static PbLi at 700-800°C during 1000 h. The results of these tests are presented.
Iron-base alloys are the leading candidate structural material for first-wall and blanket applications in near-term fusion devices, but their long-term viability to reliably function in the harsh fusion nuclear environment remains to be established. Helium produced by transmutation reactions interacts with microstructural features such as pre-existing dislocations, martensite lath boundaries, precipitate interfaces, and vacancy clusters. Helium accumulation can lead to hardening and embrittlement, bubble formation that may exacerbate swelling, and premature creep-rupture due to cavity formation at grain boundaries. An investigation of helium on damage evolution under neutron irradiation of PM2000, an oxide dispersion strengthened (ODS) ferritic alloy, was performed. While the aluminum content of PM2000 precludes it for fusion first-wall applications, the different grain structure and ODS size distribution provides a useful basis for comparing helium and dpa effects in ODS steels with nanostructured ferritic alloys. The study was done in the High Flux Isotope Reactor using the in situ helium injection (ISHI) technique. Under mixed spectrum neutron irradiation, one ISHI approach is to use a thin nickel-bearing layer applied to the surface of a TEM disc to produce high-energy alpha particles via a two-step thermal neutron reaction sequence. Helium is injected to a uniform depth of several microns. ISHI enables exploring the effects of helium on microstructure development at fusion relevant helium-to-displacement damage ratios. Microstructural and micro-chemical evolution was characterized using a suite of transmission electron microscopy techniques. The ISHI technique allows direct comparisons of neutron-damaged regions with and without high concentrations of helium. The microstructure evolution observed includes formation of dislocation loops and associated helium bubbles, precipitation of a variety of phases, amorphization of the Al2YO3 oxides, which also variously contained internal voids, and several manifestations of solute segregation. High concentrations of helium had a significant effect on many of these diverse phenomena.
O3C.1

ITER Neutral Beam Remote Cutting & Welding Development

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It is recognized that ITER will be the first nuclear installation where welding and cutting of pipes are performed routinely under Remote Handling conditions. Remote pipe maintenance tooling has been developed for JET, but conditions were such that manual deployment was permitted. Ultra-high vacuum class welding and cutting are highly skilled tasks and demand the precise control of parameters such as component geometry, joint fit-up and tool placement to give just some examples. Remote deployment of tools, necessary due to the hazardous radioactive environment, implies limited dexterity, limited vision and reduced manoeuvrability compared to manual deployment by a skilled human operator. Special considerations in the design of remote handling cutting/welding tooling must therefore be made together with rigorous testing in order to ensure the consistent creation of the optimum joint. Maintenance of the pipes is critical for ensuring availability of the Heating Neutral Beam and the Diagnostic Neutral Beam, and so in turn the availability of one of the principal heating systems supporting the ITER machine operation. The implications of a failure in either the tooling or the finished joint are serious; this together with the first of a kind deployment of such tooling by remote handling merits a significant R&D activity. This paper details the R&D activities undertaken to prove the principle of pipe maintenance by Remote Handling means in the Neutral Beam Cell. The two main objectives of this R&D activity were to: Develop prototype proof of principle pipe maintenance tools. Evaluate the tools and welded samples produced through parametric testing. This paper describes the process and outcomes through which the requirements for such prototypes and tests have been identified, the description and justification for design choices that were made, and the outcomes of the tests and recommendations for future development.
O3C.2

Overview of remote handling technologies development in China

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Before join ITER project fusion technologies development in China are focus on fusion device and plasma operation related. Components on fusion device installed, removed and maintained by personnel. Robotic technologies are never applied for fusion. China joined ITER from 2004. Scientists and engineers are involved in ITER related study and technologies development. Remote handling systems are important systems for ITER project. Chinese engineers were involved in ITER transfer cask conceptual design and learnt much from all ITER remote handling systems. With more and more Chinese scientist and engineers involved in ITER project more and more budget from the government give very strong support to develop fusion reactor related technologies both for ITER and domestic fusion reactor plan. China Fusion Engineering Test Reactor (CFETR) is China own fusion reactor plan. The mission is develop tritium self-sufficient and duty time reach more than 30% fusion reactor. Remote handling systems for CFETR should be covered all radiative components maintenance and exchange. Principle prototypes were developed to demonstrate robotic system can be applied for fusion components handling, such as for in vessel viewing, blanket modules handling, port plugs handling, components transportation, radiation resistant, reactor components and remote handling system campatible design, and so on. The presentation will present all the development, testing results and future plan of remote handling in China.
Remote handling of DEMO breeder blanket segments: Blanket transporter conceptual studies

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As part of the programme to create a viable concept design for the Eurofusion DEMO powerplant, RACE is developing a concept design for the remote maintenance system. Within the DEMO tokamak, breeding blankets will require periodic replacement. In the current DEMO design this replacement will 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 DEMO RM project to date has developed concepts for the blanket replacement process. Using a systems engineering based approach to understand the requirements for the blanket replacement activity, a key system, carrying a high level of technical risk, has been identified: the blanket transporter. The blanket transporter will be required to manoeuvre the blanket segments between the mounts and fixations within the vacuum vessel, and a position that will allow them to be lifted vertically through the upper port. This paper outlines a conceptual study to develop a feasible design for the blanket transporter. Requirements were obtained via functional analysis and CAD based kinematic analysis of the breeder blanket replacement. These requirements were used to develop a number of concepts for the main kinematic mechanism. Evaluation these lead to down selection of two concepts for further development. The proposed concepts demonstrate the potential for developing and integrating a number of technologies within the blanket transporter to produce a feasible engineering design that could be used to validate the blanket replacement strategy and hence viability of the DEMO concept. 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.
Assessment of navigation technologies for autonomous vehicles in nuclear fusion facilities

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Nuclear power plants require periodically maintenance, including the remote handling operations of transportation performed by automated guided vehicles (AGV). The navigation system becomes a key issue given the safety constrains of the heavy load to be transported in the complex scenarios, such as the reactor building. This work presents well-known and mature navigation technologies used by AGV in industry. A critical assessment is also presented regarding the performance of these technologies against the specific operational requirements and safety demonstration in the framework of fusion facilities (e.g. ITER and DEMO). The navigation technologies are based in two concepts: a physical path (e.g. wire/inductive guidance, optical line guidance and magnetic tape guidance) and a virtual path (e.g. laser based, motion capture, inertial, magnetic-gyro) to be followed by the AGV during the operations of transportation. Given the costly consequences of a failure, a solution for the navigation shall include more than a single navigation system and, hence, different candidate solutions combining different technologies are proposed and evaluated. All candidate solutions comprise a primary and a secondary navigation system, where the primary is self-sufficient for, at least, all nominal operations, while the secondary is self-sufficient for both nominal and non-nominal operations, e.g. recovery and rescue. The trade-off analysis included a broad range of criteria, organized in the following categories: technical feasibility, robustness against radiation and residual magnetic fields, availability, cost, replacement ability and use of commercial off-the-shelf. Each one of the primary and secondary provide redundancy, such that individual sub-system failure does not compromise navigation. The secondary navigation system, which will serve as back-up to the primary navigation, may be used to estimate the localization of any other vehicle (e.g. rescue vehicle). The assessment process results in a preferred solution, contributing to resilience to unexpected failures and long-term sustainability of the navigation system.
Response of the imaging cameras to hard radiation during JET operation

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In magnetic fusion devices of the next generation such as ITER, high neutron and gamma-ray yields could be detrimental to the applied diagnostic equipment such as video imaging systems as well as to electronic components of machine control systems. Semiconductors devices are particularly sensitive to the radiation, both ionizing (formation of traps at the Si/SiO2 interface with energy levels within the silicon bandgap) and non-ionizing (displacement damage effects). Defects degrade the performance of CCD image sensors by increasing the average dark-current and dark-current non-uniformity, which result in the appearance of individual pixels with high dark-currents (“hot-pixels”). The main objective of this contribution is to summarize the experience of camera operation acquired during the D-D experimental campaigns on the JET machine. The analysis of the radiation damage of imaging systems is based on all different types of analogue/digital cameras with uncooled as well as actively cooled image sensors in the VIS/NIR/MWIR spectral ranges. MCNP code has been used to determine the neutron fluence and energy distribution at different camera locations in JET. An explicit correlation between the sensor damage and the neutron fluence has been observed. Sensors show an increased dark-current and increased numbers of hot-pixels. Uncooled cameras must be replaced once per year after exposure to a neutron fluence of \(1.9-3.2 \times 10^{12} \text{neutrons/cm}^2\). Such levels of fluence will be reached after \(\approx 14-22\) ELMy H-mode pulses (reference #74176(4.5MA/3.6T, PNBI=25MW/tNBI=6s) during the future D-T campaign. Furthermore, dynamical noise seen as a random pattern of bright pixels was observed in the presence of hard radiation (neutrons and gammas). Failure of the digital electronics inside the cameras as well as of industrial controllers is observed beyond a neutron fluence of about \(4 \times 10^{9}\) neutrons/cm². The impact of hard radiation on the different types of electronics and possible application of cameras during future D-T campaign will be discussed.
This paper describes the preliminary RAMI analysis for the ITER Low Field Side Collective Thomson Scattering (LFS CTS) system based on its preliminary architecture achieved at the System Level Design. The benefits and challenges involved in a RAMI analysis since the front end of the design process of the system are discussed together with the methodology pursued. The Functional Analysis, developed both at system and sub-system level, are the major inputs for the RAMI analysis. This study includes the Failure Mode, Effects and Criticality Analysis (FMECA) and the Reliability Block Diagram (RBD) of the system. Criticality charts are developed to highlight the risk levels of the different failure modes, with regards to their probability of occurrence and effects on the availability of the ITER machine. Mitigation actions are proposed in order to reduce these risk levels in case of impact in the ITER operation. The FMECA analysis has identified two components of the system whose failure will have impact in the ITER operation: the cooling system and the evacuated waveguide in primary Vacuum. The RBD analysis shows that the initial (before mitigation) availability results are 33% obtained for the system and 96% for the ITER operation, when all components are considered to be in series. An independent analysis has been developed to assess the reliability and availability of the system: the receiver transmission lines are in m-out-of-7 parallel reliability-wise relationship and the remaining components in series. When a high level of redundancy in the receiver transmission line is considered the availability of the system is slightly above 70%. However for lower levels of redundancy the availability is clearly damaged and can be as low as 33% if all transmission lines have to be in an operational condition for the system to be considered as available.
A control and data acquisition platform for critical systems

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Abstract: The increasingly complex Physics experiments demand innovative digital Instrumen-
tation for critical Measurement and Control functions. Requested system capabilities are, at
least: high reliability, availability, maintainability, synchronized real-time high throughput data
processing and compatibility to established Standards. Some of the methods that help attaining
those capabilities are:

Redundancy mechanisms; Independent Hardware monitoring; Hardware description methods and
database; Hotplug/Hotswap mechanisms; Multi-channel real-time data flow with embedded timing
and units/error tagging; Centralized software/firmware updating methods; “Universal” common
device driver layer supporting the above mechanisms/methods; “Universal” common Network
interface layer abstracting the specific networks used (Ethernet, ExpressFabric, Infiniband . . .);
Real-time, high-throughput data processing mechanism, with a common interface to the specific
processors used (CPU, GPU, FPGA), the Universal device driver and the Network interface; High
energy radiation resilience methods.

IST/IPFN keeps a continuous development of these methods on an ATCA/PCIe standard
platform, targeting the upcoming ITER experiment and implemented in a number of current
Fusion experiments (ITER, JET, W7-X, ASDEX-U, COMPASS, ISTTOK, TCA-BR). This
communication gives an overview of the work performed and planned, thus complementing the
communications in this conference that present some of the detailed developments.
First direct comparative test of single crystal rhodium and molybdenum mirrors for ITER diagnostics

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All optical and laser diagnostics in ITER will use mirrors to observe the plasma radiation. In the severe ITER environment mirrors may become contaminated with plasma impurities hampering the performance of corresponding diagnostics. To counteract the mirror contamination, an in-situ mirror cleaning is proposed, which relies on ion sputtering the contaminants together with affected mirror material. However, such a cleaning introduces high demands on reflectivity and on sputter resistance of mirror materials. Previous research demonstrated the decisive advantages of single crystal (SC) molybdenum (Mo) under erosion conditions over polycrystalline concepts. Until now the production of rhodium (Rh) mirrors with an excellent reflectivity was limited to polycrystalline thin coatings due to high cost of rhodium and design challenges. Recently, the first single crystal rhodium mirrors became available and tests have been started at the Forschungszentrum Jülich. In a direct comparative test two SC Rh mirrors and two SC Mo mirrors were exposed under identical conditions in steady-state helium plasmas in the linear plasma device PSI 2. The energy of impinging He-ions was \( \sim 100 \) eV well matching conditions expected in the in-situ cleaning system in ITER. The temperature of the mirrors was \( \sim 300^{\circ}C \), the total fluence was \( 1.9 \times 10^{21} \) ion/cm\(^2\). During exposure molybdenum mirrors lost \( 450-600 \) nm of their material due to sputtering. Rhodium mirrors lost more than \( 1 \) \( \mu \)m. Exposure impact on mirrors corresponded to 50-100 cleaning cycles, thus addressing the entire mirror lifetime in ITER. Nevertheless, rhodium mirrors have preserved their specular reflectivity, showing the maximum degradation of less than 7% at 250 nm. The diffuse reflectivity was preserved. Molybdenum mirrors demonstrated moderate decrease of specular reflectivity of 12%-25%. Since the moderate degradation corresponds to the entire service life of the mirror, the obtained results open new perspectives for the use of single crystals in ITER diagnostics.
O4B.1

Potential approach of IR-analysis for HHF quality assessment of ITER tungsten divertor targets

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Plasma-facing units equipped with tungsten (W) monoblock geometry are employed at the vertical targets of the ITER divertor. This contribution discusses a statistical approach for high heat flux (HHF) tests as potential quality assessment of the ITER divertor additional to the quality assurance performed by the manufacturer during the manufacturing. The IR analysis of the local temperature evolution of W blocks during the first 100 cycles at 10MW/m\textsuperscript{2} could be statistically assessed. This would allow an assessment of the industrially manufactured plasma facing units, equipped with roughly 300,000 monoblocks with reasonable HHF test effort. We discuss a possible approach on the basis of the HHF test strategy of W7-X divertor manufacturing \cite{1}. Ten monoblock mock-ups with different surface machining and varying geometries were loaded with 100 cycles at 10MW/m\textsuperscript{2} in the test facility GLADIS. The surface temperature evolution was monitored with both, two-colour pyrometry and IR imaging. While the pyrometer data showed reasonable temperatures in a good agreement with 3D-FE modelling, the raw temperature IR data continuously decreased with cycle number. Temperature differences up to 200K between pyrometer and IR camera data were measured. This effect originated from removal of surface impurities and was confirmed by EDX analysis. The monoblock geometry as well as the crucial emissivity of W require a careful evaluation of the proposed method. At least two parameters describe the position and extension of a bonding defect between cooling-tube and monoblock, the circumferential position $\theta$ and the extension $\Delta\theta$. According to the 3D-FE modelling of a monoblock with a large defect $\Delta\theta$\textsuperscript{30$^\circ$}, a local surface temperature increase between 30 and 100K has to be measured. We discuss a possible correction method of the IR-data. Possible assessment criteria based on these temperature measurements will be presented. \cite{1} H.Greuner et al., FED 88 (2013) 581-584
O4B.2

Fabrication and acceptance of ITER vertical target divertor full scale plasma facing units

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ENEA and Ansaldo Nucleare S.p.A. (ANN) have been deeply involved in the European International Thermonuclear Experimental Reactor (ITER) development activities for the manufacturing of the inner vertical target (IVT) plasma-facing components of the ITER divertor. During normal operation the heat flux deposited on the bottom segment of divertor is 5-10 MW/m\textsuperscript{2} but the capability to remove up to 20 MW/m\textsuperscript{2} during transient events of 10 seconds must also be demonstrated. This component has to be manufactured by using armour and cooling pipe materials defined by ITER. The physical properties of these materials prevent the use of standard joining techniques. In order to overcome this difficulty, ENEA has set up and widely tested a manufacturing process, titled Hot Radial Pressing (HRP), suitable for the construction of these components. The last challenge is now to fabricate, by means of the new HRP facility, a full scale prototype of the IVT for the final qualification, that is the scope of the contract F4E-OPE-138 in which ENEA-ANN are now involved. The tolerances and acceptance criteria of the IVT plasma facing units (PFU) are fixed by ITER/F4E and are very tight. The objective of manufacturing a PFU that satisfies these requirements is an ambitious target. The final acceptance control to check the component compliance with the acceptance criteria is performed by ultrasonic water gap technique. For this purpose a new equipment suitable for the final control of PFUs by ultrasonic was developed and qualified in ENEA. The paper reports a description of the innovative ultrasonic equipment together with the dimensional check results of the fabricated full tungsten full scale PFUs.
DiMES PMI research at DIII-D in support of ITER and beyond

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An overview of recent Plasma-Material Interactions (PMI) research at DIII-D tokamak using the Divertor Material Evaluation Station (DiMES) is presented. The DiMES manipulator allows exposing material samples in the lower divertor of DIII-D under well-diagnosed ITER-relevant plasma conditions. Plasma parameters during the exposures are characterized by the extensive diagnostic suite including a number of spectroscopic diagnostics, Langmuir probes, IR imaging, and Divertor Thomson Scattering. Post-mortem measurements of net erosion/deposition on the samples are done by Ion Beam Analysis, and results are modelled by REDEP/WBC and ERO codes with plasma background parameters derived from OEDGE/DIVIMP modelling based on experimental inputs. This presentation highlights the following key findings: (i) demonstration of strong reduction of net compared to gross erosion of molybdenum and tungsten by short-scale redeposition, in good agreement with REDEP/WBC modelling; (ii) demonstration of good survival of tungsten nano-structures (W-fuzz) under helium plasma exposures and reduction of gross erosion rate from the fuzz surface compared to solid W surface; (iii) studies of inter-ELM versus intra-ELM gross W erosion rates, showing that for attached divertor conditions inter-ELM erosion is dominant at the strike point, while further into the scrape-off-layer, intra-ELM erosion dominates; (iv) studies of active control of sheath conditions by external electric biasing, demonstrating suppression of Mo erosion with positive biasing, as predicted by ERO modelling; (v) formation of in-situ carbon coating by local methane injection leading to suppression of molybdenum erosion, in agreement with ERO modelling; (vi) results of dust remobilization tests from W substrate, showing that overheating of the dust particles by the plasma contact leads to enhanced adhesion to the substrate. *Supported under US DOE DE-FG02-07ER54917a, DE-AC05-06OR23100b, DE-AC05-00OR22725c, DE-FC02-04ER54698e, DE-AC52-07NA27344g, DE-AC04-94AL85000h, Collaborative Research Opportunities Grant from National Sciences and Engineering Research Council of Canadaf.
HELCZA – High Heat Flux Test Facility for Testing ITER EU First Wall Components

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The ITER first wall panels are exposed directly to thermonuclear plasma and must extract heat loads of about 2 MW/m² (EU) to 4.7 MW/m² (RF + CN). The panels are qualified through high heat flux cyclic testing before the installation in ITER. Initially the first wall panel prototypes will undergo full-power tests, this will be followed by the pre-series panels and finally the series panels. The experimental complex HELCZA has been completed and is entering the commissioning phase (written first half of 2016). HELCZA will provide a cyclic heating of the ITER EU first wall panels with a heat flux in the multi-MW/m² range (the facility is capable to reach the GW/m² scale) using an 800 kW electron beam. The test area in HELCZA is about 3 m². The electron beam gun electromagnetic system provides a beam scanning frequency of 20 kHz at the primary deflection angle up to ±40 degrees. A secondary deflection system on the vessel can be used to ensure perpendicular beam incidence to the surface of the panel. HELCZA is equipped with a 3D kinematic system for panel tilting, which allows the incidence angle of the beam relative to the plate to be chosen. The test facility provides for thermo-hydraulic and infrared measurement. The tested panels are cooled by demineralised water at an inlet temperature of 70°C and at a pressure of 4 MPa. However, operational conditions of the HELCZA facility enable the temperature of the cooling water to be set between 25°C and 320°C, within a water pressure range between 1 – 15 MPa. The cooling system provides an optimal flow rate between 0 m³/h and 40 m³/h whatever the pressure. The opinions expressed are those of the CVR’s only and do not represent Fusion for Energy’s official position.
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Abstract: Fusion energy becomes essential to solve the energy problem with the increase of energy demands. Although the recent studies of fusion energy have demonstrated the feasibility of fusion power, it commonly realizes that more hard work is needed on neutronics and safety before real application of fusion energy. A high intensity D-T fusion neutron generator is keenly needed for the research and development (R&D) of fusion technology. However the intensity of D-T neutron generators currently on operation around the world is lower than $10^{13}$n/s, which is severely restricting the research capability. The Institute of Nuclear Energy Safety Technology (INEST), Chinese Academy of Sciences (CAS) has launched the High Intensity fusion Neutron Generator (HINEG) project to develop an accelerator-based D-T fusion neutron generator with the neutron yield higher than $10^{14}$-10$^{15}$ n/s. The R&D of HINEG includes two phases: HINEG-I and HINEG-II. HINEG-I is designed to generate both the steady beam and pulsed beam, and has been completed and commissioning since the end of 2015 with the D-T fusion neutron yield of up to $10^{12}$ n/s. HINEG-II aims at a high neutron yield of $10^{14}$-10$^{15}$ n/s neutrons via high speed rotating tritium target system and high intensity ion source. HINEG can be used for research of fusion nuclear technology and safety including the validation of neutronics method and software, radiation shielding and protection, mechanism of materials activation and radiation damage as well as neutronics performance of components. Its application can also be extended to nuclear medicine, radiotherapy, neutron imaging and other nuclear technology applications. This contribution will summarize all the latest progress and future plans for the R&D of HINEG.

*Corresponding Author, Email: yican.wu@fds.org.cn Keywords: D-T Fusion; Neutron Source; High Thermal Power Tritium Target; High Intensity Beam Accelerator
O4C.2

Resolving Safety Issues for a Demonstration Fusion Power Plant

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As part of the conceptual design studies for a European DEMO, a programme of safety studies and analyses is performed, intended to help guide the design process by assessing the safety and environmental impact of design options under consideration. They also begin to prepare for the eventual licensing of DEMO construction and operation by a European nuclear regulator. A safety approach has been adopted that is expected to satisfy a regulator, but at this early stage it is difficult to anticipate the safety concerns that may be raised. The French nuclear safety authorities and their technical advisors are unique in having licensed the construction of a nuclear fusion facility, ITER, and have acquired expertise in examining the safety case for a fusion facility. It was therefore useful that in 2015, the technical advisors IRSN presented the safety issues that they perceive as important in a future nuclear fusion facility such as DEMO [1]. These include the need to remove decay heat following an accidental loss of cooling or during the removal and transportation of blanket modules from the tokamak, the minimization of personnel exposure to ionizing radiation, the comprehensive identification of postulated accident scenarios, including some hazards different or additional to those encountered in ITER, the environmental release of gaseous tritium during normal operation, and the management of radioactive waste, particularly where contaminated with tritium. This paper will explain how these issues are being addressed in the safety programme for a European DEMO, and are taken into account in the design from the beginning of its conception.

Qualification of MELCOR and RELAP5 nodalization models for EU HCPB and HCLL TBS accident analyses

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‘Fusion for Energy’ (F4E) is designing, developing, and implementing the European Helium-Cooled Lithium-Lead (HCLL) and Helium-Cooled Pebble-Bed (HCPB) Test Blanket Systems (TBSs) for ITER. Safety demonstration is an essential element for the integration of these TBSs into ITER and accident analysis is one of its critical components. The F4E, Amec Foster Wheeler and INL comprehensive methodology for fusion breeding blanket accident analysis, published last year, consists of several phases. The methodology starts with the selection of reference accident scenarios, the development of detailed accident analysis specifications and the assessment of analysis codes. Models of each TBS are then constructed using the selected codes (MELCOR 1.8.5 and RELAP5-3D for the HCLL TBS) and modelling approaches. The models are qualified according to a test matrix including comparison with TBM finite element design analyses, code-to-code comparisons (between the MELCOR 1.8.5 and RELAP5-3D models) for both TBS normal operation and transient cases, and sensitivity studies for accident scenarios. The qualification test cases that are executed gradually move from models of separate systems to complete TBS models, and from the simulation of steady-state and normal plasma pulse operation to consideration of power excursions, operational transients and accident events. Finally, both of the qualified models are used to analyse a selected accident scenario (a 32 hour loss-of-offsite power) together with sensitivity studies dedicated to the evaluation of uncertainties. This step completes the qualification process. The impact of uncertainties associated with the accident analyses is also addressed to provide confidence in the level of conservatism in the results. Following an expert review of areas of uncertainty (including phenomena identification and ranking table (PIRT)) a gradual approach to uncertainty assessment has been adapted. The results obtained in the qualification of the EU HCLL and HCPB TBS models and their uncertainty evaluation will be reported in the paper.
The ASTEC code has been recently extended to address the analysis of the main design basis accident scenarios in fusion installations, more particularly in the ITER facility. Current efforts are focused on loss of coolant accidents (LOCA) because a strong reactivity between beryllium toxic dust and steam leading to possible formation of gaseous beryllium oxide, hydroxide and hydride during the transient is expected. The accidental scenario presently considered is a wet bypass between the first and second confinement barriers. This corresponds to a multiple failure of wall cooling loops inside the vacuum vessel coupled with a failure of both windows in a heating line. In our previous source term evaluations for different LOCA scenario, the presence of tritiated beryllium dust in modules at the transient end was put in evidence. With the presently considered scenario, due to the bypass, the transport of toxic dust towards the galleries becomes possible. In this study, aerosol transport modelling has been improved, notably the material properties, i.e. density, thermodynamic data and thermal conductivity which are shown to significantly impact the aerosol behaviour. The isotopic effect and neutron damage are also taken into account. The properties of condensed beryllium hydride and hydroxide have been reviewed with a critical assessment of experimental literature data and completed by computational quantum mechanical modelling. The influence of this improvement on the transport, speciation and location is shown. A special attention is paid to compounds formed with tritium and beryllium. Moreover a parametric study on sensitive material properties is presented in order to evaluate their potential impact on the source term.
Current status of the EU DEMO Project on the inner fuel cycle systems

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In the framework of the EUROfusion DEMO Programme and its work package Tritium-Matter Injection-Vacuum (TFV), the EU is preparing the conceptual design of the inner fuel cycle of a pulsed fusion DEMO. This contribution presents the current status of the project, addresses the most demanding challenges and shows first results. The project was started in 2014. The first one and a half years were devoted to the development and application of a systems engineering approach to find the most suitable concept to meet the functional needs, in particular the necessity to minimise the tritium inventory. As a result, a novel architecture was derived, which features an innermost direct internal recycle loop as shortcut between the divertor pumping and the pellet injectors, a second loop inside the tritium plant without complete separation of the hydrogen isotopes, and the classical outer loop with full separation of the different hydrogen isotopologues. Based on that, individual R&D programmes have been launched in the different sub-areas of tritium plant inner and outer systems, core fuelling and gas injection, and vacuum pumping (NBI and torus). In a recent exercise, the R&D plan has been extended in full detail until 2020. This paper will start with an outline of the R&D strategy and a discussion on the uncertainty in the requirements and assumed input parameters from other DEMO work packages. It will then highlight first results for all sub-systems. This includes modelling examples for isotope separation in the tritium plant, open loop modelling of pellet injection, and simulation of metal foil and diffusion vacuum pumps. For tritium and matter injection, experimental plans will be presented, whereas in the area of vacuum pump development results from validation experiments will be reported. Finally, first results of an integrated fuel cycle simulation exercise will be shown.
Fusion Reactor Start-up without an External Tritium Source

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Although the D-T reaction is the most promising for fusion and is widely promoted, the amounts of tritium necessary to provide a sustainable fuel supply do not exist naturally. Besides the tritium must be self-sufficient operating a reactor, the initial fuel loading to start up any large-scale D-T fusion reactor remains a significant issue. We have examined the feasibility of starting a reactor from the D-D reactor. There are two likely D-D fusion reaction channels, 1) D+D→T+p, and 2) D+D→He3+n. The tritium can be generated via the reaction channel ‘1)’ and the 2.4MeV neutrons from ‘2)’ react with lithium-6 in the breeding blanket to produce more tritium to be fed back into plasma fuel. Quantitative evaluations are conducted to parametrically assess the feasibility and suitability of this approach to FPP reactors, including the economic impact of operation without net electricity generation. The results suggest that D-D operation may be required for ~1 week to 2 months to accumulate sufficient start-up tritium to launch a tritium seeded D-D dominant plasma operation. The fusion power increases from 9.3MW (~2.53MW from secondary D-T fusion reactions), for pure D-D fusion, to 44MW (~37.5MW from D-T fusion reactions), for the mixed fuel fusion operation with 1% tritium injected into the plasma core. As soon as the operation starts to consume tritium in fusion reactions, the tritium accumulation will be largely dependent to the surplus tritium bred in the breeding blanket. The tritium fractional burn-up is a critical factor determining the length of time taken to accumulate sufficient tritium for 50:50% D-T operation. The time taken to accumulate sufficient tritium for 1 day operation would be more than 2 years for 5% burn-up but reduce to 8 months for 20% burn-up fraction assuming 2.4GW D-T fusion power and TBR of 1.2.
O5A.3

Recent developments in the TRIPOLI-4® Monte-Carlo code for fusion applications

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TRIPOLI-4® is a 3D continuous-energy Monte-Carlo particle transport code, developed by CEA, and devoted to shielding, reactor physics, criticality safety and nuclear instrumentation. TRIPOLI-4® is currently able to simulate four kinds of particles: Neutrons from 20 MeV down to $10^{-5}$-5 eV, Photons from 50 MeV down to 1 keV, Electrons and positrons from 100 MeV down to 1 keV.

The TRIPOLI-4® version 9 (released in 2013) includes some features especially conceived for the fusion community, such as the possibility to define particle sources by means of subroutines (without recompiling the code) and the addition of the torus volume in the TRIPOLI-4® geometry. External tools are also available that allow converting a CAD model into a TRIPOLI-4® geometry: MCAM developed by FDS Team and McCad developed by KIT. For the TRIPOLI-4® version 10 (released for in December 2015), a new option has been added concerning the energy interpolation mode used for the neutron exit energy distribution. This option offers the users the opportunity to choose between the raw evaluation and a more physical treatment of these data (unit-base interpolation). Thus, it is now possible to quantify the impact of such an interpolation on high energy neutron and coupled neutron-gamma sources for fusion analyses. Moreover, an activation scheme is being developed in TRIPOLI-4® to calculate shutdown dose rates for fusion applications. It is based on the Rigorous-Two-Step method (R2S) coupling the Monte-Carlo code TRIPOLI-4® with the depletion code MENDEL: TRIPOLI-4® Monte-Carlo code performs a neutron transport calculation in order to compute the flux in each region susceptible to produce decay gammas. MENDEL depletion code computes the nuclide inventories and the decay photon sources for each region based on the neutron fluxes previously calculated by TRIPOLI-4®. TRIPOLI-4® transports the decay photons and computes the dose rates induced in each region of interest.
**O5A.4**

**Assessment of HCLL-TBM optimum welding sequence scenario to minimize welding distortions**

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This work was performed by CEA within the framework of one specific contract concerning the development for ITER of manufacturing procedures for the industrial ATMOSTAT (ALCEN group) and Fusion For Energy (F4E). The HCLL-TBM (Helium Cooled Lithium Lead Test Blanket Module) box assembly development implies the welding development of the following components: the Box and the Stiffening Grid (SG) made of vertical and horizontal Stiffening Plates (noted respectively v-SP and h-SP). This multi-chamber box structure in EUROFER97 steel is made of plates cooled by meandering multiple channels where circulates pressurized helium. For the assembly of these components, characterized by numerous multipass welds, Gas Tungsten Arc Welding (GTAW) is envisaged as reference process. Moreover, the TBM has large dimensions and thin plates which makes it very sensitive to welding distortions and is problematic regarding the assembly feasibility and compliance with geometric tolerances. This paper presents the numerical simulation and experimental work performed to optimize the v-SP to box assembly sequence, which is the most critical assembly regarding distortions, in order to minimize welding distortions. One of the technical lock of this study is high calculation times needed for this big component which implies to set up a simplified welding simulation method. The study is composed of two main phases: an experimental-numerical study of a T-joint fillet mock-up GTAW welding (representative of TBM welds) used to develop the welding procedure and to validate a simplified simulation methods (chosen by testing and comparing two methods); and a numerical optimization of the v-SP to box welding via the simplified method validated in the first phase. The calculation and comparison of three different v-SP to box welding sequences allowed us to identify the best sequence regarding welding distortions to apply experimentally in the future.
Post-irradiation high heat flux investigation of plasma facing components

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To qualify new plasma facing materials (PFM) and to evaluate the high heat flux performance under ITER or DEMO relevant loading conditions, extensive High Heat Flux (HHF) testing is indispensable. This includes performance tests under cyclic stationary thermal loads and screening of different material candidates under relevant transients such as Edge Localized Modes (ELMs) with high pulse numbers. In addition, these thermal load tests have to be performed under conditions which also account for damaging influences such as the degradation due to plasma exposure by hydrogen and helium ions and the impact of energetic neutrons. Quasi-stationary and transient events performed under these harsh conditions have strong impact on the lifetime of the wall armour and – in the worst case – can result in a catastrophic failure of the PFCs. Therefore, performance tests on specific PFCs in future fusion devices are not limited to thermal fatigue or thermal shock induced experiments under mitigated ELMs, but must also include neutron irradiation induced material degradation and the impact of hydrogen and helium induced effects (such as embrittlement, blistering and fuzz formation) to allow reliable predictions on the lifetime of PFCs. Due to the lack of a powerful 14 MeV neutron source, irradiation experiments with ITER relevant neutron doses are performed in fission type material test reactors. Plasma facing materials and components have been and are being irradiated up to ITER-relevant fluences of approx. 1 dpa at temperatures ranging from 200 to 700°C. Post irradiation examinations (PIE) include a detailed analysis of the heat flux performance under the above mentioned loading scenarios as well as the neutron induced degradation of thermal and mechanical properties. A new test facility at Forschungszentrum Jülich will also allow to study synergistic effects (combined thermal loads, plasma exposure and neutron irradiation effects).
O5B.2

Radio-frequency design of a lower hybrid slotted waveguide antenna

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This paper presents the Radio-Frequency (RF) design of a new type of slow-wave Lower Hybrid Current Drive (LHCD) launcher, based on the Slotted Waveguide Antenna (SWA) concept, which is particularly attractive for the use in future magnetic fusion reactors. When compared to conventional LHCD slow-wave launchers, SWA are less obstructive, allow an “off-port” extension of the launcher and are particularly convenient for wave-injection from non-equatorial ports, which can be beneficial for the current drive efficiency. It also allows launching the waves into the plasma at low power density. There is no apparent restriction to combine SWA with the inherently load-resilient multi-junctions, or to adapt them with the ITER-relevant passive-active concepts. A low power prototype of a Slotted Waveguide Antenna has been designed for the COMPASS tokamak at IPP-Prague. The RF design of this low power prototype is based on hybrid electromagnetic calculations, combining Finite-Element-Method (FEM), Mode Matching (MM) and RF network computations. FEM calculations are performed using ANSYS Electronics Desktop® for characterizing the waveguide vacuum components, while ALOHA [1], which is based on MM calculations, is used for computing the coupling of radiating slots into the magnetized plasma. Contrary to the simulations of conventional grill launchers, the scattering matrices for both vacuum and plasma sides include higher order modes in the SWA case. The SIDON [2] network solver is then used to combine the Generalized Scattering Matrices (GSM) resulting from FEM and MM calculations. The paper presents in addition the design of the array feeding circuit, combining waveguide and low power coaxial technology. [1] J. Hillairet, et al, Nucl. Fusion 50 (2010) 125010. [2] W. Helou, et al., Proc. 21st Topical Conf. on RF Power in Plasmas, AIP Conf. Proc. 1689 (2015) 070004.
O5B.3

Design considerations for future DEMO gyrotrons: a review on related gyrotron activities within EUROfusion

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Long term options for a steady state DEMO may require the availability of gyrotrons with an operating frequency above 200 GHz together with an RF output power of significantly more than 1 MW and a total gyrotron efficiency higher than 60 %. Fast frequency tuning in steps of around 2-3 GHz will be needed for control of plasma stability. Multi-purpose operation at leaps of about 30 – 40 GHz (e.g. at 136/170/204/238 GHz) might be considered for plasma start-up, heating and current drive at different operation scenarios. The combination of all those requirements clearly challenges present day technological limits. Focusing on named aspects offers the possibilities for the development of advanced technologies. It will provide the answer to the fundamental question of maximum achievable operating frequency and corresponding output power and efficiency. At the same time, it helps to find answers to questions related to fundamental operation stability and manufacturability of RF sources operating at lower frequencies also. The R&D work within the EUROfusion WP HCD EC Gyrotron R&D and Advanced Developments (AD) is focusing on named targets. In particular, a center frequency of around 240 GHz is under investigation, considering the requirements for “multi-purpose” and “fast frequency step-tunable” operation also. In this frame, significant investments in advanced Brewster-angle window technology is considered. Coaxial-cavity gyrotron technology, and, as a possible fall-back solution, the conventional-cavity technology are under investigation. Both technologies are studied with regards to maximum achievable output power versus efficiency, operation stability and tolerances. Concerning the coaxial-cavity technology, an additional experimental investigation shall verify the predicted operation capabilities. Different promising concepts for multi-stage depressed collectors (MSDC) are under investigation. The research and development are completed by advancing the simulation and test tools capabilities significantly. A comprehensive review on the research and development will be provided in this presentation.
Multi-section Traveling Wave Antenna for heating of large machines as DEMO

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The main advantages of Ion Cyclotron Resonance Heating and Current Drive (ICRH&CD) are its ability to achieve power deposition in the centre of the plasma column without any density limit along with direct heating of plasma ions. The challenge is then to couple large amount of power through the plasma boundary, where an evanesence layer has to be crossed, without exceeding the voltage standoff at the antenna. A solution presently considered is the reduction of the power density by means of antennas distributed all along the wall of the machine. In reference \cite{1} we have shown that a suitable launcher can be constituted by sections of Travelling Wave Antenna (TWA) mounted in resonant ring systems. They are launching a traveling wave in one direction along the structure that leaks its energy to the plasma and it is refilled periodically by generators. Each section is constituted by a series of equidistant mutually coupled grounded straps aligned in the poloidal direction which radiates its power to the plasma proportionally to the total strap number divided by their inter-strap distance. Due to the large number of radiating elements, the launched power spectrum is very selective. A detailed discussion on the multi-section antenna is made in view of its test on a mock-up. We study the influence of its geometrical parameters on its response along with the influence of the periodicity of the sections and the feedings. This extends the work done in \cite{2}. The aim is to prepare for a proof-of-concept system to be tested in an operating tokamak machine. \cite{1} R. Ragona and A. Messiaen, submitted for publication (2016). \cite{2} A. Messiaen and R. Ragona, EPS Conf. 2016 This work has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053.
First Operational Phase of the Superconducting Magnet System of Wendelstein 7-X

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The Wendelstein 7-X stellarator (W7-X), one of the largest stellarator fusion experiments, is presently in the first operational phase at the Max Planck Institute for Plasma Physics in Greifswald, Germany. The W7-X shall prove the reactor relevance of the optimized stellarator concept. To confine $30m^3$ plasma the W7-X machine has a superconducting magnet system with 50 non-planar and 20 planar coils grouped in five equal modules, electrically connected in seven circuits with 10 coils of each type. The connections between the coils are made by superconducting bus bars using the same NbTi Cable-in-Conduit Conductor as used for the superconducting coils. Specially developed high temperature superconducting current leads feed the current into the cryostat vacuum by bridging the temperature gradient from room temperature down to the 4 K level. Seven power supplies provide individual currents in the seven circuits. The quench detection system checks permanently the superconducting system regarding the occurrence of a quench. In case of a quench, the magnet safety system has to be activated and a set of switches lead the current into dump resistors. The magnet commissioning was successfully performed until mid of 2015 with tests of the complete magnet system functionality needed for plasma operation, at a magnetic field of 2.5 T. The first operational phase started mid of December 2015 with He plasma heated by the ECRH (Electron Cyclotron Resonance Heating) system followed by H2 plasma in January 2016. The superconducting coils and their nonlinear support structure are equipped with a large set of mechanical sensors e.g. strain gauges, contact and distance measuring sensors. For these sensors an online monitoring is established to detect any deviations from the behavior as predicted. The paper will present the experiences from the operation of the superconducting magnets during the first plasma operational phase.
Development and Commissioning of the Wendelstein 7-X Safety Control System

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The Wendelstein 7-X safety control system is one of the main central control entities and ensures personal safety and investment protection. Its proper definition and setup has been a major precondition for the operation permit by the authorities and was inspected by external reviewers several times. The safety control systems has a distributed architecture comprising of the central safety system with safety signal interfaces attached to components like the cryo plant, superconducting magnets, heating systems and many more. These components have either a local safety control system or just actuators that are directly controlled by the safety signals. The development and commissioning process has been established according to the engineering standard for functional safety in industrial processes (IEC 60511). The development processes with respect to the safety instrumented system starts with the safety requirements specification, in which safety instrumented functions are defined. On the requirements level, the unified modelling language and finite state machine simulations have been used for confirmation of the desired functionality. The software runs on a fault tolerant Siemens PLC with distributed interface Profibus-Safe devices and has been implemented with the Siemens PCS7 programming environment. The commissioning has been done in two steps, one stage for the evacuation and cool-down of the cryostat and the final stage for the preparation of the first plasma. The safety programs have been verified for both development stages and finally validated against the safety instrumentation functions. Throughout the development and commission process the personal safety had to be ensured while deploying new safety program releases and performing integration tests of attached components.
An optimized upper divertor with divertor-coils to study enhanced divertor configurations in ASDEX Upgrade

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ASDEX Upgrade came into operation in 1991. It was designed as a tokamak with reactor relevant shaping. The coil and control system allows to operate in lower single null (LSN), double null (DN) or upper single null (USN) with up to 1.6 MA plasma current and an initially open divertor configuration. Divertor enhancements were concentrated on the lower divertor that was finally transferred to a solid tungsten divertor with vertical target plates and a large flexibility for magnetic configurations. Consequently, the physics program of ASDEX Upgrade is concentrated on LSN magnetic configurations using this optimized lower divertor. The extension of the physics program towards the investigation of advanced divertor configurations requires a new divertor design, the installation of in-vessel coils and of a cryopump. This modification will be done in the upper divertor, keeping the flexibility for physics investigation in the lower divertor as in the past. We will present the conceptual design for the upper divertor structure with embedded divertor coils fed with up to 50 kAt, the hardening of the upper divertor target structure to cope with the high heating power of ASDEX Upgrade without leading edge effects and finally the cryopump for an effective density control in the upper divertor. Possible coil designs and the forces acting onto the coil support structure and the vacuum vessel will be discussed in detail. The design of the cryopump is based on the existing design for the cryopump in the lower divertor but has to be modified to be fitted behind the upper inner divertor. Hardening of the upper divertor will be done by installing well aligned flat or roof like targets and a minimum gap size between targets. Such a design would keep the option to operate with both directions for the helicity of the magnetic configuration.
Optimal current profile control for enhanced repeatability of L-mode and H-mode discharges in DIII-D

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To collect meaningful experimental data, it is necessary to maintain consistent operating conditions in the tokamak plasma across repeated discharges. Presently, the desired plasma formation conditions, such as the shape of the plasma current profile, are achieved in a trial and error fashion, which can be a lengthy, wasteful process. In this work, model-based control techniques including optimal feedforward control and linearized feedback control are used to obtain a desired current profile at a specified time in low-confinement-mode (L-mode) as well as high-confinement-mode (H-mode) discharges. The evolution of the current profile is closely related to the evolution of the poloidal magnetic flux profile, which can be properly modeled in a first-principles manner by a nonlinear partial differential equation (PDE) referred to as the magnetic flux diffusion equation (MDE). Simplified, control-oriented formulations of the magnetic diffusion equation have already been developed for the DIII-D tokamak for both L-mode and H-mode discharges. In both cases, the control-oriented models combine the MDE with physics-based correlations for the electron temperature, plasma resistivity, and non-inductive current drive sources including neutral beam injection (NBI), electron cyclotron current drive (ECCD), and bootstrap current. With the use of these models, an open-loop control problem, i.e. an actuator trajectory optimization problem, is formulated to find a feasible path from the expected initial condition to the desired target. The result comprises a sequence of feedforward (open-loop) control requests and a corresponding state evolution from the initial condition to the desired target. On top of this optimal feedforward control sequence an optimal state feedback (closed-loop) controller based on a linearized model is added to track the desired state evolution. Experimental evidence of the effectiveness of the control approach in reaching the targets and facilitating repeatability between discharges is presented.
A tee or an elbow behaves very differently from a straight pipe in resisting bending moment. When a straight pipe is bent, its cross section remains circular and the stresses increase linearly with distance from the neutral axis. However, when an elbow or a tee is bent, its cross section gets deformed into an oval shape. This geometrical deformity results in increased stresses, which are accounted for by using a factor called Stress Intensification Factor (SIF). The SIFs are specified in applicable Codes and Standards and they can be used with some limitations. ASME B31.3 Appendix D provides SIFs obtained from tests on full size branch connections for tees. But in applications where the branch size is much smaller than the main run pipe (diameter ratio d/D < 0.5), use of stress intensification factor from codes is too conservative. This can cause overestimation of stresses and unnecessary design modifications in piping systems. In such cases, an improved and reduced SIF can be used, if more applicable data available. ITER secondary cooling systems (Component Cooling Water System, Chilled Water System and Heat Rejection System) have numerous interfaces with client components located in Tokamak Complex and other auxiliary buildings across the site. These systems include large amount of piping with diameters ranging from 0.15 to 2m and countless intersections. Precise SIF values need to be used for stress analysis of these piping under service levels C&D considering the huge amount of static and dynamic loads to arrive at an optimized design and layout. In this paper, a finite element approach has been adopted to find out SIF values for branch-offs as well as header pipes at intersections with varying branch sizes to compare them with the code specified values and their advantages in stress analyses of ITER CCWS, CHWS and HRS piping.
P1.002

Design and Structural Analysis of the Upending Tool

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The purpose of the Upending Tool (UT) is to upend the vacuum vessel (VV) 40-degree sectors and the toroidal field coils (TFC) from horizontal delivery orientations to vertical assembly orientations. According to the ITER assembly procedure, this upending operation is carried out by four hooks of the tokamak crane. And the VV and TFC which are upended with UT are transfer from the UT to sector sub-assembly tool (SSAT) for sector sub-assembly. For this reason, the UT is classified as a lifting accessory in the ITER load specification of PBS22. This paper describes the UT design and upending sequence by crane with its components (VV and TFC) and presents the analysis results performed for the final design to verify the structural integrity of UT. The analysis cases for this verification are composed of the horizontal supporting on the floor, horizontal lifting ($0^\circ$), $30^\circ$ tilting, $45^\circ$ tilting, $60^\circ$ tilting, vertical lifting ($90^\circ$), and vertical supporting on the floor. And, in this analysis, the mass and inertial moment of the components (VV and TFC) are reflected into point masses.
Structural analysis for final design of ITER sector sub-assembly tool

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The Sector Sub-assembly Tool is a special tool for assembly of ITER Tokamak and is used to sub-assemble the 40° Tokamak sector which consists of vacuum vessel sector, vacuum vessel thermal shield sector and two toroidal field coils. The sector assembled in the assembly building is a basic and fundamental unit for the construction of the ITER Tokamak. Therefore, the design and structural integrity of the Sector Sub-assembly Tool is very important. To assess the design and structural integrity of the Sector Sub-assembly tool, the structural analyses have been performed under the Category I and II load combination according to load specifications. The Category I load combination is including two kinds of loads. One is the normal operation case and another is the factor case. The factor case includes the dynamic amplification factor and design factor. The Category II load combination is including the normal case and the seismic load. The seismic load is SL-1 event. The analysis code is used ANSYS code. The results of the structural analyses show that the design and the structural integrity of the Sector Sub-assembly Tool meet the requirements. This paper provides briefly the result of final design for the Sector Sub-assembly Tool.
P1.004

SHe cooling performance evaluation of 600 W Helium Refrigerator/liquefier with variable temperature supplies

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A 600 W He refrigerator/liquefier with variable temperature supplies was constructed in National Institute for Fusion Science (NIFS) and its operation is started. Several cool-downs of large sized superconductors and magnets, such as a conductor of ITER TF coil and a JT-60SA superconducting coil, will be performed. The cooling performance is confirmed to meet its specifications. Two dummy heat loads were used to measure cooling capacities in the supercritical and the gas phases. Cooling capacities at 20 K and 40 K are apparently higher than the designed values. On the other hand, the cooling performance of Supercritical He (SHe) could not be estimated to be more than the requirement which is 350W cooling capacity at 4.55 K with 50 g/s at the beginning of the commissioning test. Therefore, the performance had to be proved to meet its design specifications. The gross heat load should have been evaluated from the enthalpy rise at the dummy heat load and the mass flow rate through it. It is the sum of the heater power and heat inleak at the dummy load. However the flow meter equipped in it was uncalibrated at that point. It means that the gross heat load and the mass flow rate had been uncertain. Then we applied a new method to evaluate them. The correlation between the net heater power of the dummy heat load and the resulting SHe enthalpy rise was systematically investigated under a certain mass flow rate and temperature. Finally, the calibration of the mass flow meter and the evaluation of the heat inleak were succeeded. The SHe cooling performance of 422 W at 4.42 K with 51.3 g/s has been proved. In this paper, we describe the method of the SHe cooling capacity measurement and mass flow meter calibration.
This paper describes the analysis performed for the final design review of the ITER Gas Distribution System (GDS) manifolds to verify the system structural integrity. The GDS manifolds, which consist of Gas Fuelling (GF) manifold and Neutral Beam (NB) manifold, are complex combination pipes, of which gas supply lines and evacuation line are enclosed in a guard pipe. Based on the loading conditions and safety requirements, four categories of events and distinct criteria levels and corresponding load categories and service limits have been defined first, and then the structural failure modes including the plastic collapse, ratcheting, local failure and buckling have been verified. The simplified pipe element models have been used in linear elastic static analysis and spectrum analysis, and the typical sections for buckling analyses have been discretized by solid element considering material nonlinearity. The analysis results showed that both the GF and the NB manifolds are safe under all loading cases and the structural integrity requirements against all the failure modes are well satisfied. The GDS manifold design is robust with enough safety margins.
Detailed design of ITER CCWS, CHWS and HRS: Challenges experienced and their solutions
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While the decisive feat of any concept is ‘successful implementable design’, the process of converting the concept into practically executable design is critical and challenging. It is usual to initiate any design on the basis of challenges visible during the conceptualization, as no project can really be a repeat of another. However, during conceptual design phase, it may not be possible to include all key interfaces and practical issues as they emerge progressively with the progress of design. Unforeseen challenges emerged and, along with the foreseen ones, tested the progress of detailed design of ITER Component Cooling Water System (CCWS), Chilled Water System (CHWS) and Heat Rejection System (HRS). ITER being ‘one-of-a-kind’ project, has unique and very conservative design requirements. While trying to meet these requirements through system design, revisiting of those requirements was essential so as to complete the design within the constraints. In addition, as some of the auxiliary client system designs progressed through prototyping, new or revised requirements were identified. Since ITER is an experimental facility, accommodating the provisions like early operation, partial operation, capacity augmentation influenced the progress of detailed design. In a project like ITER, where the interfaces are complex and rigid, the development and implementation of the system design was challenging, given the fact that multiple collaborating agencies were performing concurrent engineering. This paper captures the challenges faced during the detailed design phase of ITER CCWS, CHWS and HRS and describes the solutions that were eventually found in addressing those challenges while maintaining compliance to the technical requirements.
Mechanical design solution for cold water basin of ITER Heat rejection system

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ITER is an experimental fusion reactor being constructed in south of France which will demonstrate the scientific and technological capability in the direction of future commercial fusion power plant. The enormous amount of heat generated from the experimental reactor (mainly from the In-vessel components of Tokamak and its auxiliary systems) shall be removed by the Primary, Secondary and Tertiary cooling systems respectively. The Primary Heat transfer Systems (PHTS) receive heat from core components of fusion reactor and reject heat to the Secondary heat transfer systems which is in turn removed by the Tertiary cooling system, also identified as Heat Rejection System (HRS). The HRS finally rejects heat to the atmosphere. The heat rejection system is designed to remove a peak heat load of approximately 1100 MW with the help of a cooling tower with a capacity of approximately 510 MW, cold water and hot water basins, vertical turbine pumps and interconnected piping. The cold water basin serves a dual purpose. It provides the main support to the cooling tower (comprising of 10 cells each of 16m square FRP construction) and also contains a large body of water, having a capacity of 10000 cubic meters. The cold water basin which accommodates set of vertical turbine pumps submerged in the water, is designed economically to suit the site requirement within the limited space available as a site constraint. The purpose of this paper is to describe the practical mechanical design solution for the ITER cooling tower cold water basin, initiated during the preliminary design phase after considering site and cost constraints and the need for a reliable and environmentally friendly design. This paper also reveals the challenges in carrying out general arrangement and layout design and describes the viable mechanical design solution for the cold water basin of Heat rejection system.
P1.008

Preliminary design of local magnetic shield for solenoid valve in FPSS of ITER gis

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The function of Gas Injection System (GIS), in ITER machine, is to deliver the fuelling and impurity gases into the torus. As an important sub-system of GIS, Fusion Power Shut-down System (FPSS) provides the function of emergency shut down for torus safety. The assessment of magnetic field in Tokamak building shows that a high stray field will exist in port cells during burning plasma operation and its maximum value at FPSS box location can reach 0.205 T in ‘Full Operation Scenario 17MA’ case\(^2\). Thus, shielding for magnetic sensitive component inside FPSS box becomes an important issue to be solved in FPSS design phase. Inside the FPSS box, only the SV, used to actuate the pneumatic isolation valve, is susceptible to magnetic field. Experiments show that the maximum allowable magnetic field for the SV is 0.013 T considering a margin of 3 dB\(^3\). Therefore, the objective of the shielding design is to ensure the residual field lower than 0.013 T within the shield space. A soft iron local shield is considered from engineering design viewpoint since only one SV needs to be shielded from stray magnetic field. Half open structure with inner dimension of 120mm×115mm×60mm and 30mm thickness is designed to keep the accessibility and maintainability of the SV. The layout is carefully designed to avoid the magnetic interference as much as possible. The magnetostatic analysis by ANSYS APDL software package indicates that the maximum residual field will be lower than 0.003 T under any direction of poloidal field of 0.205 T. At least 50% design margin is left and more details will be present in the paper later.
First evaluation of cryogenic performance of Wendelstein 7-X cryostat

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The first cool down of the stellarator fusion experiment Wendelstein 7-X was achieved within 4 weeks in March 2015. A helium refrigerator with a cooling power of 7 kW at 4.5 K was used to cool down 456 tons of cold mass. The Outer Vessel (OV) of the cryostat contains 70 superconducting coils that are threaded over the twisted Plasma Vessel (PV). These coils are attached to a massive support structure. Both, coils and structure are cooled down to 4 - 5 K. 14 high temperature superconducting current leads connect the warm power cables to the cold superconducting bus bars which are joined with the coil terminals. Heat radiation from the warm surfaces to the cold structures is prevented by a thermal radiation shield around the PV, the OV and 254 ports. The ports allow access to the PV from outside the cryostat and contain supply/return lines and plasma diagnostics. More than 1200 temperature sensors of different types are attached to cryostat components and give information on the temperature of components. The paper presents the behavior and the analysis of the cryogenic components inside the cryostat. Heat loads and temperature distribution on the thermal shield are described and compared with design calculations. Heat loads on the cold support structure and on the cold coils system with and without current in the coils are discussed. The impact of loss of cooling for 2 days on coils and structure temperatures is explained and the resulting pressure rise in the helium manifolds over time is presented. The analysis demonstrates that achieved temperatures and measured heat loads allow a safe operation of the superconducting coil system. The cryostat fulfills its requirements.
Refrigerator operation during commissioning and first plasma operations of Wendelstein 7-X

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On 13th February 2015 began the cool-down of about 450 tons cold mass of Wendelstein 7-X i.e. 70 superconducting magnets, 14 currents leads, massive support structure and the thermal shield, enclosed within a vacuum vessel of about 15.4 m outer diameter. After a smooth cool-down, the temperatures around 5 K, within the so called Short Standby Mode with the thermal shield return temperature < 70 K, were achieved on 9th March 2015. Since then, for about 1 year, these components are kept cold at different temperatures within various operating modes of helium refrigerator. The refrigerator is sized to provide an equivalent cooling power of 7 kW at 4.5 K with the help of 2 screw compressors consuming 1.6 MW electrical power, 7 turbines, 2 cold compressor and 4 cold circulators. In the Standard Mode (SM), the supply temperatures were reduced to 3.9 K using the cold compressor and the mass flows were increased to 200 g/s through the magnets and 300 g/s via the magnet casings and structure with the help of cold circulators. The commissioning of magnets with currents up to 13.4 kA and the plasma operations with about 2.5 T magnetic field (with different currents in non planar and planar magnets) were carried out in the SM. Over the Christmas vacations, the temperatures were raised to 100 K within so called Long Standby Mode. The refrigerator operation was optimized for each of these modes by achieving the guarantee values of mass flow, pressures and temperatures. The details of the refrigerator operation at different modes, operation conditions of main equipments, failures, repairs etc. shall be presented in the paper.
Influence of deviations in the coil geometry on Wendelstein 7-X plasma equilibrium properties

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Wendelstein 7-X (W7-X), went into operation in December 2015 at the Max-Planck-Institut für Plasmaphysik in Greifswald, Germany, is a modular advanced stellarator with a magnetic field optimized for good plasma confinement and stability [1]. The magnet system of W7-X consists of 70 superconducting coils - ten non-planar and four planar in each out of five modules of the machine. Preliminary simulations of the plasma equilibrium properties were performed taking into account as-designed shapes and positions of W7-X coils. In these calculations each coil was represented by one filament subdivided in 96 cross-sections. As-built positions of coil filaments might considerably deviate from their as-designed values due to manufacturing and positioning tolerances or due to coil deformations under different kinds of loads. Step-by-step evaluation of the magnet system geometry during the machine construction included measurements of the winding pack geometry in eight points characterising deviations of each coil cross-section caused by the fabrication process as well as the tracking of coil positions during main assembly steps [2]. In addition finite element calculations were performed with help of the 360° ANSYS Global Model in order to define possible coil deformations under the dead-weight, cool-down and electromagnetic loads in different operation regimes. This presentation shows the comparative analysis of different types of deviations in the coil geometry and discusses their influence on plasma equilibrium properties, which allows to define a reliable basis for the detailed analysis of different diagnostic data. References [1] GRIEGER, G., et al., “Modular stellarator reactors and plans for Wendelstein 7–X”, Fusion Technol. 21 (1992) 1767-1778 [2] BRAEUER, T., et al., “Interaction of Metrology and Assembly at W7-X”, IEEE Transactions on Plasma Science, Vol. 42, No. 7, (2014).
A software eco-system for the integrated design of W7-X

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Wendelstein 7-X (W7-X) is a fusion device of the stellarator type with optimized magnetic field geometry and superconducting coils. The scientific goals of W7-X are to confirm the predicted improvement of the plasma confinement and to demonstrate the technical suitability of such a device as a fusion reactor. It is undergoing its first operation phase at the Max Planck Institute for Plasma Physics (IPP) in Greifswald, Germany. The Design Engineering division is responsible for the design tasks and for the integration of all the components in the Torus Hall. The complex management of space, the numerous designers operating simultaneously, the maturity gap of the designs pose acute concurrent engineering issues. Originally used to manage the space reservation, the PLM solution ENOVIA SmarTeam is now the backbone of the all the design tasks and at the heart of an eco-system that helps the designers to fulfill their tasks efficiently and reduce the risk of iterations. This eco-system is a set of software tools developed internally that interact with SmarTeam. The collision analysis, the delivery process and the global CAD mockup generation are some of the tasks with are speed up and secured through this eco-system. In this paper the context and the challenges of the design process will be introduced. The role and implementation of the software tools will be presented and the benefits will be discussed through examples.
Monitoring of W7-X Cryostat Commissioning with Cryostat System FE Model

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The Wendelstein 7-X stellarator started its first operational phase in October 2015 at the Max-Planck-Institute for Plasma Physics in Greifswald with the goal to verify that a stellarator magnetic confinement concept is a viable option for a fusion power plant. The main components of the W7-X cryostat system are the plasma vessel (PV), outer vessel (OV), 254 ports, thermal insulation, vessel supports and the machine base. The main task of the cryostat system is to provide an insulating vacuum for the cryogenic magnet system, UHV conditions within the PV and to provide external access to the PV through ports for diagnostic-, supply- and heating systems. The updated finite element (FE) Global Model of the Cryostat System (GMCS) has continued to be used for predicting and assessing the behavior of W7-X as measured during its commissioning and operational phase. The measurements with strain, temperature and displacement sensors as positioned on the OV, Ports, PV and its supports form the basis of the cryostat system monitoring. After successful evacuation of the OV [1] commissioning continued and in 2015 the PV has been evacuated and baked for the first time. The measurements show good correspondence with the predictions of the GMCS and allowed for continuation of the commissioning. This paper gives an overview of analyses performed with the GMCS in support of cryostat commissioning and operation. In addition, the assessments performed for optional PV position adjustment is presented. A PV adjustment might be required in case plasma operation reveals problems with the plasma heat load distribution on critical in-vessel components. [1] P. van Eeten et al., “Features and analyses of W7-X cryostat system FE model”, Fusion Engineering and Design, Volumes 96–97, October 2015, Pages 369–372
P1.014

Tokamak COMPASS-Upgrade support structure study

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This contribution describes the electromagnetic and structural analysis of the new structural design of the COMPASS-U tokamak. The electromagnetic calculations solve force effects on tokamak coils using ANSYS Maxwell 3D code. The calculations were performed for three different combinations of excited coils and for two different plasma positions. The structural analysis was performed then using ANSYS Mechanical. Electromagnetic set-up considered the central solenoid and poloidal and toroidal field coils, positioned in a cylindrical computational domain. The analysis also includes the plasma column, modelled as a conductive object with a defined electric current. The electromagnetic analysis solves the static magnetic field and forces that act on the tokamak coils. The support structure, acting forces, and the coils are considered in the structure analysis. FEM model is built to describe the support structure, the stiffness of the tokamak structure was not influenced by the modelling method. The forces calculated in the electromagnetic analysis and the gravitational acceleration are used as the load in the static structural analysis which solves displacement field, stress, and behaviour of the contact interface between the structural parts the COMPASS-U tokamak. The results of the structural analysis give similar displacement field and stress in all investigated variants. The analysis demonstrates significant torsional displacement of the central parts and high loads due to high pressures that are generated in the central solenoid. The perimeter support shows high stiffness. The results serve to the team at Institute of Plasma Physics of the Czech Academy of Science to validate and upgrade the structural design of COMPASS-U. The results will be also used to suggest suitable structure design measures to reduce torsional displacements in the tokamak centre and thus to reduce the high mechanical stress in that part of the tokamak.
Integration design platform of CFETR

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The design of the Chinese Fusion Engineering Test Reactor (CFETR) must integrate a great number of working documents and data from many groups, and distribute these materials to everyone in time, therefore, the parallel design work in different places could be properly managed, and the schedule, as well as the cost, could be ensured. An integration design platform has been built with this demand; all design groups will receive design requirements and share their progress on this platform, even directly work on it. In the 1st stage of construction, for CFETR conceptual design, two main components are set up: 1) the project document management, including the project description, the sub-system requirement and conceptual design reports; 2) the engineering CAD/CAE design, including the 3D design tool like CATIA and multi-physics analysis tool like ANSYS; all the data are stored in a tree structure and a library is defined for easy search and summary, every single document and CAD/CAE file is linked to corresponding design branches to be traceable. A comprehensive system design code is being developed on this platform, which extends to multi-dimensional physical and engineering design of the CFETR; it has modular structure, and different functional modules are seamlessly connected. An advanced GPU-based cloud server cluster provides a remote design environment, which brings the virtual workstations to the designer’s desktop with local experience, and then the data is generated and stored directly on CFETR central servers. In the future, more components will be added to this platform including the physical design branch. The detail of integration design platform will be presented in this conference.
Study of current ramp up for the China Fusion Engineering Test Reactor

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The ramp up scenario design, which considers of both physics and engineering constrains, plays an important part in fusion device design. The Tokamak Simulation Code (TSC), coupling with some auxiliary heating codes, has been implemented in the CFETR system code to construct the workflow of the CFETR ramp up scenario designs. In this workflow, the CFETR geometric construction design and some preliminary physics parameters are firstly prepared as the input of the 1.5D TSC computation. Then different discharge scheme are obtained by TSC with coupling the poloidal field systems controller with passive structure, free boundary equilibrium solvers and the auxiliary heating sources. The electron cyclotron (EC) and neutral beam (NB) sources are used to heat the plasma and drive the toroidal plasma current form the output of ONETWO code. Based on the time-dependent plasma current density distribution evolutions simulated by TSC, ANSYS is used to analysis the 3D electromagnetic loads and stresses in the main structural components, which are used to estimate the safety operation of CFETR. In this presentation, some scenarios with different ramp rates or heating schemes are also presented.
Comparison of HESEL SOL turbulence simulations with BES measurements on EAST

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The HESEL code has been used to simulate scrape-off-layer (SOL) electrostatic interchange-driven low-frequency turbulence in various EAST tokamak discharges [1]. The recently installed Lithium Beam Emission Spectroscopy (LiBES) diagnostic system on EAST provides well resolved non-intrusive 2D measurements of SOL turbulence [2]. This paper presents results of comparison of statistical properties of simulated and measured SOL turbulence making use of the RENATE synthetic beam emission diagnostic [3]. It has been found that the blob dynamics involves spatial scales that are strongly affected by the limited spatial resolution of the LiBES system, thus detailed modelling of the LiBES diagnostic is essential. For the purpose of comparison, discharges with optimal diagnostic performance have been selected from the latest campaign. HESEL simulations were carried out using the geometry of a 2D slab perpendicular to the magnetic field lines at the outboard midplane, and provide time dependent density and temperature fields. For the purpose of interfacing with the RENATE synthetic diagnostic, these 2D fields were toroidally extended along magnetic field lines to provide 3D fluctuating density and temperature fields. These were then used for the detailed 3D simulation of the BES diagnostic system with RENATE at every time instance, including effects resulting from a spatially extended beam and arrangement of the viewing optical system, as well as the effect of atomic physics processes in the beam. Comparison of simulated and measured SOL turbulence was based mostly on spectral and correlation methods that are typically used in experiments. [1] N. Yan et al. 2013 PPCF 55 115007[2] To be published in Rev. Sci. Instrum.[3] D. Guszejnov et al. 2012 RSI 83 113-501
Properties of plasma injected in open magnetic trap from independent UHF source

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Various ways of filling the open magnetic trap with plasma are used in different experiments on study of plasma in order to develop methods of plasma heating and confinement, to study the interaction of electromagnetic waves with magnetoeactive plasma etc. Among all existing methods the ultra high frequency (UHF) contactless methods are used frequently. We have proposed the method of filling the open magnetic trap (with a homogeneous and mirror configuration of magnetic field) by plasma injection along the magnetic field from the independent stationary UHF source separated from the trap. In this paper stationary UHF plasma source, its characteristics and possibility of filling open magnetic trap with plasma injected from the source are investigated. Plasma is created in the UHF source at the frequency 2400 MHz (150W) in the electron cyclotron resonance (ECR) regime under working gas pressure $10^{-5}$÷$10^{-2}$ Torr. By changing discharge conditions one can modify the injected plasma density from $10^{8}$ to $10^{12}$ cm$^{-3}$, at the temperature $T_e = 2\div 3$ eV. The possibility of efficient plasma injection from the source into the open magnetic trap with uniform field is shown experimentally. Properties of plasma in the trap are presented under various experimental conditions. It turned out that plasma lifetime in the trap is determined by classical mechanism of particle escape at the expense of collisions. At fixed value of magnetic field in the trap plasma lifetime practically does not change with variation of neutral gas pressure and reaches the value $4\times10^{-3}$ s at magnetic field strength in the trap equal to 1600 Oe. A set of experimental data gives the possibility to conclude that such method of filling the open magnetic trap with plasma can be successfully used in various physical experiments.
P1.019

Measurement of flow velocity during natural convection in nanofluids

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Increased cooling performance is eagerly required by the cutting edge engineering and industrial technology. Nanofluids have attracted considerable interest due to their potential to enhance the thermal performance of conventional heat transfer fluids. However, heat transfer in nanofluids is a controversial research theme as there is yet no conclusive answer to explain the underlying heat transfer mechanisms. The purpose of this study is to investigate the physics behind the heat transfer behavior of Al2O3 – deionised (DI) H2O nanofluids under natural convection. A high spatial resolution flow visualisation method (Particle Image Velocimetry - PIV) is employed in dilute nanofluids inside a classical Rayleigh-Benard configuration with appropriate optical access. The resulting mean and instantaneous velocity and flow structures of nanofluids and their overall heat transfer performance are compared with those of pure DI water, under a broad range of Rayleigh numbers. In this way, the possible modification of flow structures due to the addition of nanoparticles will be evaluated and its potential influence on the heat transfer rate in nanofluids assessed. Additional comparisons between current experiments and numerical studies of different modelling approaches and boundary conditions will be reported to assess the accuracy of the numerical and analytical tools. This paper aims to identify the contribution of the suspended nanoparticles on the heat and mass transfer mechanisms in low flow velocity applications, such as natural convection. In addition, the outcome of the current research is a first step towards the evaluation of the applicability of nanofluids in applications where more complex heat transfer modes, namely boiling and Critical Heat Flux, are involved that are of great importance for the cooling of Fusion reactors.
Fabrication of DLC cone for fast ignition experiment

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Fast ignition is one of the proposed ways to achieve high fusion energy gain in inertial fusion research. This scheme has an advantage that requirements of laser power and implosion process for ignition are not strict compared to that in central ignition. For a successful ignition, it is necessary to transport the energy of hot electrons to the imploded core effectively. Recently, it is found that hot electrons were diverged more than expected. In addition, it is concerned that hot electrons are scattered by high-Z plasma generated from gold cone target. This may cause the drop of the energy coupling of the heating laser to hot electrons. Therefore, low-Z materials, such as diamond like carbon (DLC) and aluminum, are drawing attention as cone materials. However, it is very difficult to deposit thick DLC layer for making a stand-alone DLC cone because of its strong residual stress. In this study, we tried various preparation conditions for thick DLC layers and studied its characteristics. DLC layer was prepared on metal conical bars by using plasma–based ion implantation and deposition system. Acetylene gas or toluene vapor was used as a source. It was found that the toluene vapor had an advantage in thick layer deposition because of its high deposition rate. It was found that rf pulse power, negative bias pulse and gas pressure affected deposition rates. These DLC layers showed SP³ rich property in Near edge X-ray absorption fine structures (NEXAFS) spectra. Moreover, we improved the DLC cone fabrication process (cutting and etching). Based on these results, we succeeded in making stand alone DLC cones more easily.
Installation and site testing of the SPIDER Ion Source and Extraction Power Supplies

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SPIDER experiment, currently under construction at the Neutral Beam Test Facility (NBTF) in Padua, Italy, is a full-size prototype of the ion source for the ITER Neutral Beam (NB) injectors part of the ITER project. The Ion Source and Extraction Power Supplies (ISEPS) for SPIDER are supplied by OCEM Energy Technology s.r.l. (OCEM) under a procurement contract with Fusion for Energy (F4E) covering also the units required for MITICA and ITER injectors. The installation of SPIDER ISEPS started in June 2015 and was completed in September 2015. Functional checks started thereafetrer and power testing in January 2016. The formal Site Acceptance Tests (SAT), witnessed by F4E, Consorzio RFX and the ITER Organization were successfully completed in April 2016. ISEPS, with an overall power rating of 5 MVA, form a heterogeneous set of items, ranging from power transformers, medium voltage power distribution equipment at 6.6 kV to solid state power converters and including four 1 MHz radiofrequency generators of 200 kW output power. Both high voltage, down to -12 kV and high current, up to 5 kA, power supplies are present. The paper presents the main features of the SPIDER ISEPS installation inside the SPIDER HV Deck, focussing in particular on non-standard technical solutions like the cable terminations on the HV Deck incoming power lines. The arrangements are described for fully checking the integration of the ISEPS local control system with the SPIDER control system are described. Details are given on non-standard site tests verifying immunity to electromagnetic interference (EMI) and the behaviour of the control electronics during fast discharges of electrostatic energy, specific to this application. Performance of ISEPS during the SAT is described and commented against the specifications, with emphasis on demonstration of the 1 hour full load capability of the radiofrequency generator.
The transmission line for the SPIDER experiment: from design to installation

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SPIDER (Source for the Production of Ions of Deuterium Extracted from RF plasma) is the 100keV Ion Source Test facility (presently under construction in the Neutral Beam Test Facility at Consorzio RFX premises, in Padua, Italy) representing the full scale prototype of the Ion Source (IS) for the ITER 1 MeV Neutral Beam Injector (NBI). SPIDER Ion Source, polarized at -100kVdc Power Supply, is meant to produce Deuterium or Hydrogen negative ions which, after being extracted by the extraction grid, are accelerated up to ground potential. The required Ion Source and the Extraction Grid Power Supplies (ISEPS) system and the associated diagnostics need to be hosted inside a -100kVdc air-insulated Faraday cage, called High Voltage Deck (HVD), while a High Voltage Transmission Line (TL) transmits the power and signal conductors from the ISEPS to the Ion Source. An air insulated design of the TL, duly screened against Electromagnetic Interferences (EMI) produced by the frequent IS grids breakdowns was preferred to a more complex and costly Gas Insulated Line design because of the large diameter required by the TL inner conductor to host all ISEPS power and signal conductors, fibres optic and cables. The TL is procured (together with the HVD) by COELME SpA, via a procurement contract with Fusion for Energy (F4E) started mid 2013. The paper describes the construction solutions developed by the Manufacturer to meet the design indications and technical specification requirements for such unconventional device. Moreover, the results of factory type tests, carried out on a first section of the TL mid-2015 to validate the design and release the manufacturing of the overall TL, are also described. Finally, the paper reports on the on-site installation and commissioning activities, presently ongoing, up to the final acceptance foreseen within the current year.
The SPIDER Central Interlock is a centralized electronic system to coordinate the protection functions within the SPIDER experiment (Source for the Production of Ions of Deuterium Extracted from an RF plasma), i.e., the full-ion source prototype of the ITER Neutral Beam Injector. Due to the system time requirements, the SPIDER Central Interlock has been implemented by using PLCs for the slow functions (10 ms reaction time from fault detection to protection command emission) and National Instruments CompactRIO for the fast functions (10 µs from fault detection to command emission). The paper will describe in detail the system hardware and software architecture. The result of the first tests and commissioning will be reported with particular focus on the implementation, execution, and performance of the interlock fast functions.
The ITER project requires at least two Neutral Beam Injectors (NBIs), each accelerating to 1MV a 40A beam of negative deuterium ions, to deliver to the plasma a power of about 33 MW for one hour as additional heating. A full-size negative ion source (SPIDER - Source for Production of Ion of Deuterium Extracted from RF plasma) and a prototype of the whole 1 MV ITER injector (MITICA - Megavolt ITER Injector & Concept Advancement) are under construction at PRIMA a new test facility for the ITER neutral beams being built at the Consorzio RFX in Padua, Italy. In SPIDER, the source is housed in a stainless steel vessel maintained in high vacuum. A three grid system is foreseen to extract and accelerate at 100kV the beam. For this reason, the vacuum vessel hosts is equipped with three large alumina feedthroughs, ring shaped insulators (800mm in diameter) to insulate the vessel from the hydraulic and electrical services which supply the source components biased at different potentials in the range between -112 and -100kVdc. These insulators (bushings) are one of the most critical items in the SPIDER experiment and they required a particular effort in designing, manufacturing and identifying the proper test procedure. This paper gives an overview of the design, development, manufacturing activities and presents the results of the high voltage tests for the bushing acceptance.
The ITER project requires at least two Neutral Beam Injectors (NBIs), each accelerating to 1MV a 40A beam of negative deuterium ions, to deliver to the plasma a power of about 33 MW for one hour as additional heating. Since these requirements have never been experimentally met, it was recognized necessary to build-up a test facility, named PRIMA (Padova Research on ITER Megavolt Accelerator), in Italy, including both a 100 kV full-size negative ion source (SPIDER - Source for Production of Ion of Deuterium Extracted from Rf plasma) and a prototype of the whole 1 MV ITER injector (MITICA - Megavolt ITER Injector & Concept Advancement). 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, current uniformity and duration.

The SPIDER experiment needs a dedicated Cooling Plant to remove up to 10 MW heat loads applied during the one hour pulses to the beam source, the beam dump/calorimeter and the power supplies. The total water flow rate necessary to remove the heat and guarantee the optimal control of temperatures is about 150 kg/s at full operational conditions. The plant is widespread over the whole PRIMA site area and is characterized by several physical interfaces with the SPIDER experiment, the related power supplies and control system and finally with PRIMA buildings and civil plants. A well-equipped system of sensors and feed-back control system is also present to guarantee the correct performances of the plant during different operational phases. The paper deals with the installation and testing phases of the plant under execution, with particular emphasis on integration problems, electrical insulation, tests of immunity to electromagnetic interference, vibration issues, monitoring and safety issues for possible activated water and finally the acceptance tests before integrated commissioning.
The 10-MW EPSM Modulator and other KIT FULGOR Gyrotron Test Facility key components

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The construction of the new FULGOR test facility (Fusion Long Pulse Gyrotron Laboratory) at KIT is in full swing. This will significantly expand the experimental capabilities at KIT to CW tests of high power gyrotrons of up to 4 MW output power at operating frequencies up to 240 GHz. Thus, this facility will be a significant platform for the verification of the performance of current CW gyrotrons and of future generation ECRH sources as required for example for DEMO. The 90 kV / 120A CW EPSM modulator and the extension to 130 kV/120A short pulse (5ms) should become operational in 2017. Further key components, such as a 10 MW water cooling system (initially equipped for 5 MW), a superconducting 10.5 T magnet, one or two 2 MW ECRH test loads and new control and data acquisition systems for all these elements are being constructed or procured at the same time. The paper describes in some detail the High Voltage DC Power Supply (HVDCPS), a so called Enhanced Pulse Step Modulator, first proposed and built by Ampegon AG for the TRIUMF project in Canada, which will be capable of supplying single stage and multi-stage depressed collector gyrotrons with up to 4 MW RF Output power. It is for the first time that such a modulator is constructed for more than 10 MW DC output power for continuous operation. Other key components, such as a 10.5 T / 270 mm bore superconducting magnet (preferably LHe-free), the 20 kV Mains Supply, the 10 MW cooling system, the safety access system and the control and data acquisition system will be briefly outlined.
Electron Emission Under Uniform Magnetic Field of Materials For Fusion And Spatial Applications

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The power handling of RF components can be limited by a resonant process known as Multipactor effect. Multipactor can be fatal to microwave systems in space communication payloads or in experimental fusion devices. Multipactor simulations are used to predict voltage thresholds but the results highly depend on the electron emission properties of the RF components materials. Moreover, both space and fusion applications devices deal with DC magnetic fields. These are generated by permanent magnets in satellites while toroidal and poloidal coils create an intense magnetic field in fusion reactors. In order to improve understanding and predictability of multipactor effect, materials electron emission measurements under uniform DC magnetic field is carried on. An experimental test-bed has been developed to measure electron emission for fusion materials such as copper and alumina with TiN flash on its surface. Our goal is to measure the Total Electron Emission Yield (TEEY) used as input in multipactor simulations. We work under a 10⁻⁶ Pa pressure and an electron gun with an energy range between 1 eV and 2000 eV. A 41mm-diameter solenoid coil made with copper generates an uniform magnetic field perpendicular to the sample surface. TEEY measurements have been made on uncontrolled surface morphology samples to understand the influence of the magnetic field on electron emission. Then specific morphology surface samples have been used to reduce the electron emission using the electron motion induced by the uniform magnetic field. The multipactor sensitivity to these fields has been thus characterized considering samples roughness and magnetic field direction and strength. In this paper we describe the development of the test-bed which allows us to measure electron emission for fusion materials under uniform DC magnetic fields in order to improve our understanding and the predictability of multipactor effect.
High power test of a temperature controlled diplexer for electron cyclotron current drive system

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A neoclassical tearing mode (NTM) can be controlled by electron cyclotron current drive (ECCD). Up to now, ECCD with pulse modulated gyrotron operation at a duty of 50% have been done to drive current into only O-point of magnetic island of NTM. The fast directional switch have been developed for improving a stabilizing efficiency of NTM [1]. It makes the duty of ECCD system to 100% by switching beam direction for tracking the rotating magnetic island. The wideband diplexer as a fast switching device of high power millimeter wave was proposed for a step tunable gyrotron [2]. The wideband diplexer was simulated numerically, and it was tested at the frequency of 170 GHz in low power [3]. However, the rf beam could not be switched by the frequency modulated diplexer with slotted metal half mirrors in high power tests [4]. There were two main reasons of this failure. One is large Ohmic loss of metal half mirrors [5]. The other is some variations in gyrotron frequency, so that the operation frequency band of diplexer was out of the gyrotron frequency band. In this paper, the first results of the temperature controlled diplexer with sapphire half mirrors at a frequency band of 170 GHz are reported. The operation frequency of diplexer can be controlled by thermal expansion using a precision chiller, where the tunable frequency band is about 225 MHz. The switching operation is clearly observed in high power tests. References [1] W. Kasparek, et al., Fusion Sci. Technol. 52, 281(2007). [2] M. Saigusa, et al., Proc. of 13th AMPERE Toulouse, 285 (2011). [3] M. Saigusa, et al., Fusion Eng. Des., Vol. 88, 964(2013). [4] M. Saigusa, et al., 20th Topical Conf., Vol.1580, 562(2014). [5] K. Atsumi, et al., Fusion Eng. Des., Vol. 8, 2405077 (2013).
Coupled thermal-hydraulic and thermal-mechanical analysis of a 1MW gyrotron cavity cooled by mini-channels

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During operation, the resonance cavity of a high power gyrotron experiences a very large heat load (>15 MW/m\textsuperscript{2}), localized on a very short (< 1 cm) length, where any thermal deformation should be carefully controlled to guarantee the gyrotron performance. Different strategies can be considered for the removal of the heat there, among which we focus here on the use of mini-channels drilled in the annular region around the cavity. A mock-up of such a cavity has been designed, fabricated and tested at the FE200 electron beam test facility of AREVA with the double objective of checking the cooling performance and acquiring experimental data to validate/calibrate the 3D computational analysis, which was performed using the commercial software STARCCM+, especially as far as the turbulence and the boiling models are concerned. The temperature distribution on the heated surface and the temperature of the solid structure on the inner part of the mock-up assembly are measured during the tests by an infra-red camera and by a set of thermocouples, respectively, and compared to the computed values. The numerical model, after its calibration, is applied to the analysis of a 1 MW gyrotron cavity in nominal operation. The thermal behavior of the cavity under nominal heat load is computed, assuming a tentative deformation of the cavity. Then a 3D thermo-mechanical model of the cavity is developed, and simulations are performed, based on the temperature maps computed by the thermal-hydraulic analysis, to evaluate the resulting deformation of the inner cavity surface. The deformation is used in turn to re-assess the heat load coming from the electron beam in nominal operation, which becomes the input for a new iteration of the thermal-hydraulic and thermal-mechanical analysis, until the requested tolerance/accuracy has been reached.
The stabilization of appearing MHD modes (NTMs, RWMs) is a key factor in optimizing tokamak operation towards fusion power production. In NTM control, the primary actuator is a confluence of focused electromagnetic wave beams, which are generated by high-power millimetre-wave sources (gyrotrons), transferred through waveguides and injected into the plasma by a controlled electromechanical launcher. In connection to the design of controllers for the overall process, a useful theoretical model for the wave actuation should include, among others, the dynamics of the steerable launcher mirror, the motion transmission to the mirror by the associated servomotors, as well as the beam propagation in the transmission line and the plasma. In this work, we present fragments of the physics-based theoretical modelling involved in the control system setup: (a) We investigate the options for the manipulation of the geometry of the EC resonance layer, taking into account all the parameters amenable to external control (magnetic field, launcher angles, wave frequency) and their dynamic response, (b) We present a dynamic equivalent model for the electrical current vector and mechanical torque of the AC servomotors (instead of the commonly-used rms current model), (c) We study the closed-loop system error and settling time of the launcher mirror as a function of the system (plant and controller) parameters for different scenarios pursued in the experiment.
Recent results on ECR assisted plasma start-up, current drive and discharge cleaning in SST-1

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SST-1 Tokamak employs Electron Cyclotron Resonance (ECR) assisted pre-ionization as an effective support towards low loop-voltage plasma start-up at fundamental (O-mode) and second harmonic (X-mode). A 42GHz 500KW 500ms ECR source is used for this purpose. In recent experimental campaigns in SST-1, several experiments have been carried out on ECR assisted pre-ionization, plasma start-up, possible EC assisted current drives and ECR discharge cleaning. In the EC-assisted start-up experiments, 175kW power in fundamental O-mode is launched from the low field side that resulted in successful plasma start-up. With the available loop voltages, plasma current more than 100kA have been achieved with EC-assisted pre-ionization. The EC power is restricted to avoid the runaway electron generation. The EC pulse duration is varied from 80ms to 125ms. It is observed that in the initial phase of breakdown the hard X-rays appears and it subsequently reduces as the pulse progresses. This confirms good absorption of EC power at fundamental harmonic at higher density and temperature in the current ramp-up phase. The EC current drive (ECCD) effect is also observed in SST-1 as a SS304L, reflector is installed at the inboard side of tokamak to reflect the ECR beam in electron direction to assist the plasma current. The plasma discharges show around 10% contribution of ECCD. The ECR discharge cleaning is also carried out with 42GHz ECRH system at a field of 1.3T bringing the ECR layer close to inboard side near the wall. The EC power (~100kW) has been launched with pulse duration of 100 to 200ms with pulse repetition rate of 1 Hz. The H2 pressure is 1x10^-5mbar. The RGA scan shows increase of hydrogen and oxygen with ECR pulses in helium plasma confirming the wall conditioning. After the ECR cleaning, better plasma discharges are observed with higher current and duration.
Design of dual-frequency transmission lines for the ECRH systems on HL-2M

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To carry out research related to electron cyclotron waves, 6 MW ECH systems including four 105 GHz/1 MW/2 s and two 140 GHz/1 MW/3 s units will be developed on the HL-2M tokamak being built in the first stage. Dual-frequency transmission lines with same components for the 105 GHz and 140 GHz systems are designed to make the fabrication easier. The corrugated waveguides are used to ensure the bandwidth of the transmission systems. In addition, the parameters of polarizers are specially considered to obtain the required polarizations at the two different frequencies. The transmission efficiency of all the six lines is analyzed with the multimode propagation model, and also the polarization characteristics of polarizers are calculated in the two different frequencies. The results indicate that both the transmission efficiency and the polarization characteristics can meet the requirements of ECRH systems on HL-2M.
The JT-60SA tokamak is scheduled to start operations in 2019 to support the ITER experimental programme and to provide key information for the design of DEMO scenarios. The device will count on ECRH and NBI as auxiliary heating and EC operations are foreseen for EC assisted startup, EC Wall Cleaning (ECWC), bulk heating and current drive and MHD control, for example. 7 MW of total injected EC power will be available at operating frequencies: 82 GHz, 110 GHz and 138 GHz. Potentially dangerous situations that could harm tokamak structures during operations characterized by low absorption and therefore high levels of EC stray radiation cannot be a priori excluded. An estimation of the wall load under direct beam exposure and during plasma operations such as assisted breakdown, ECWC and main heating phase will be given, determined from the EC antenna main characteristics and launching geometry. A system to minimize the risk of damage to machine components would rely primarily on fast detectors to monitor the level of stray radiation at accessible (ex-vessel) lines of sight, like those provided by vacuum windows. Pyrodetectors may be suitable to monitor the fast changes in stray level, with the additional advantages of small dimensions and low volume occupation, allowing easy mounting in the proximity of windows. Though no solution could grant complete elimination of risks, local information only with in-situ detector is nevertheless beneficial, and the analysis performed aims to determine suitable positions of fast detectors and probes in the vicinity of ECRH launchers and of potentially critical beam trajectories. A proposal for the system conceptual design for JT-60SA is presented and an example of the time response and potential benefits of such a system will be given, based on experience gained on the TCV tokamak during ECWC experiments performed in preparation of JT-60SA operations.
Development of an ICRH antenna system at W7-X for plasma heating and wall conditioning

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An ICRH antenna system is developed and will be attached to W7-X for the operational phase 1.2. An antenna box with two straps with surfaces adapted to the 3d LCFS in standard magnetic configuration (m/n=5/5), is located at the low field side in the equatorial plane. The antenna system is optimised for plasma heating and wall conditioning in presence of magnetic field. Each strap is connected via coaxial lines to a generator working in the frequency range from 25 to 38 MHz. The operation of different magnetic configurations at W7-X requires radial shift of the antenna box in the plasma vessel over a distance of 350mm to achieve an optimal coupling of the RF with the plasma. Additionally each strap at one side is connected with a capacitor to optimise the resonant circuit and at the other side with the surrounding grounded antenna box. For safety reasons two air-vacuum feedthroughs in line are placed in each transmission line to minimise the risk of unintentional venting the plasma chamber. For the same reason all supplies for antenna diagnostics are passing an intermediate vacuum between plasma chamber and air. All components in the antenna head are water cooled at up to 10 bars sufficient also for wall conditioning at 150°oC. The temperatures at the antenna box and straps are measured and are leading parameters for the feedback control of the antenna position. At both toroidal sides of the antenna box gas injection pulses can be initialised in case the RF coupling with the plasma gets lost. 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 No633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.
Characterization and performance improvement of large titanium sublimation pumps in AUG and W7-X NBI

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The experimental devices ASDEX Upgrade (AUG) and Wendelstein-7X (W-7X) are both equipped with two neutral beam injectors each for plasma heating (up to 20 MW). Four large titanium sublimation pumps (TSPs) (4×1.5×0.2 m³) in each injector provide proper vacuum conditions (below 10⁻² Pa) during the 10 s beam pulse with a gas feed of up to 30 Pa·m³/s. A maximum pumping speed of up to 3000 m³/s for H₂ is obtained by frequent renewal of the Ti coating at the pump surfaces. This is achieved by ohmically heating 4-m-long hanging pairs of Ti sublimation wires above the Ti sublimation temperature (˜1800 K). Each TSP contains 27 pairs of sublimators with about 9.5 kg of Ti, theoretically allowing for over 16 h of sublimation per pair, long enough to operate the injector during regular AUG experimental campaigns for as long as 20 months. However, the injectors must be opened during this time once or twice for TSP maintenance in order to shorten the residual lengthening of the sublimators which accumulates after each thermal cycle and would result in an electrical short-circuit, hampering the operation. This residual elongation is caused by material creep of the pure tantalum core of the wires. A new type of Ti sublimation wires has been tested with a TaW-2.5%w alloy core material, less prone to material creep than pure Ta, achieving over 21 h operation without wire shortening. Additionally, tests have been performed in a dedicated test facility in order to characterize the pumping performance with a controlled H₂ gas feed for varying parameters of the sublimation cycle (e.g. power, duration). The results are used to simulate fully operative TSP systems for real NBI operation scenarios in AUG and W7-X, aimed at finding an optimal sublimation strategy.
Fabrication of the ICRF antenna of ASDEX-U device

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Abstract: Wave heating in the Ion cyclotron range of Frequencies (ICRF) has been a method of choice for plasma heating in fusion research because of its flexibility, cost effectiveness and plug-to-power efficiency. A new three-strap ICRF antenna, designed for ASDEX Upgrade, and aiming to lower RF sheath by preventing undesirable currents induced in the antenna frame, demonstrated experimentally to reduce the impurity production. It will be one of the potential and promising solutions applicable to fusion reactor. The ICRF antenna fabrication is a project to undertaken in CAS/ASIPP as one part of the international cooperation agreements, conducted jointly by IPP (Garching, Germany) and ASIPP (Hefei, China). ENEA (Italy) also fabricated components of the antenna. Based on the high requirements for the antenna, the material choice including filler material, the welding method, the measurement & inspection and heat treatment and assembly were all carefully considered and rigorously defined before fabrication. Tools of fixtures and jigs, as well as clamps were designed for manufacturing and assembly. Advanced electron beam welding (EBW) was chosen for the antenna components because of high requirements of welding quality and low deformation. Some workpieces and one mockup of central conductor were manufactured for welding test and inspection for welding qualification. In addition, heat treatment was employed to release heat stress after welding. This paper is mainly focused on the fabrication of the ASDEX Upgrade ICRF antenna components made in the framework of the CAS/ASIPP project. Based on the high accuracy required for the ICRF components, machining process is introduced, following by welding, qualification tests, assembly, scanning, measurement and metrological analysis. Keywords: ASDEX-Upgrade, ICRH, fabrication
Helium neutral beam injection into ASDEX Upgrade
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ASDEX Upgrade’s (AUG) neutral beam injection (NBI) is primarily designed for deuterium injection and delivers 20 MW heating power from two injectors with four beams each at 60 and 93 keV, respectively. As opposed to the cryosorption pumps of the JET NBI, the Ti getter pumps of the AUG NBI with a pumping speed of $\sim 3 \times 10^6$ L/s for D\textsubscript{2} do not pump helium at all, leaving only the conventional pumps with $< 6 \times 10^3$ L/s for He. This imposes constraints on the possible operation in helium. In order to prepare for AUG He plasma campaigns, serious trials to operate the AUG NBI with He began in 2014. It was found that despite the lack of high speed pumping up to two beams per injector could be operated simultaneously at reduced feed gas flow without particular restrictions on the beam-on time. For injector 1 the power per beam was limited to $\sim 560$ kW at 40 keV by the required filament current in its arc sources, while for injector 2 the limitation came from the bending magnet’s power supply that restricted the beam energy to 68 keV and the NBI power to $\sim 750$ kW per beam. Thus the maximum available NBI heating power in He amounts to 2.6 MW for 10 s. Helium neutral beam injection into plasma was first tried out in 2014 for two discharges. In 2015 a dedicated He campaign used He NBI in a total of 44 discharges. As He is almost not pumped in the injectors the neutral gas flow into the torus is comparable with the total He gas puff. The paper will discuss the operational possibilities and constraints of He NBI on AUG and give examples of the NBI heating profiles for some of the He target plasmas of 2015.
Extension of heating and pulse power capabilities at ASDEX Upgrade

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One of the biggest challenges for a fusion reactor with magnetic confinement is the controlled removal of the heating power. ASDEX Upgrade (AUG) is the leading experiment in this area and investigates integrated solutions that combine high heating power and wall materials suitable for reactors. A measure of the challenge to remove the power in the divertor region is given by the normalized output power $P_{sep}/R$. In AUG it is already 10 MW/m for 2 s, 2/3 of the value expected for a reactor and a world leading achievement. To go to even higher values for longer times, an extension of the heating power and power supply systems is required. The 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. With these measures $P_{sep}/R$ will be increased up to 15 MW/m, in the range of the values intended for ITER and DEMO, for more than 5 s. The program is split into 5 topics: - Enhancement of the ECRH to 8 MW/10 s. - Installation of 2 3-strap antennas followed by an additional 3rd generator feeding the new antennas. - Strengthening of NBI injector 1 by modern RF-sources and possibly later increase of the accelerating voltage. - Upgrade of the power supply for ohmic heating by a new high current converter. - Installation of a 90 MVAr static reactive current compensation for flywheel generator EZ4. The paper describes the motivation and program to increase the heating and pulse power capabilities of AUG towards achieving a stationary behavior of the current profile and wall-particle-inventory under reactor relevant conditions. We provide more detail on the envisioned 5 steps, the present state and findings, and future exploitation.
Simulation of burn control for DEMO using ASTRA coupled with Simulink

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DEMO is a proposed demonstration fusion power plant which is under design. Fusion power, Pfus, has to be controlled at certain level to produce sufficient net electricity. However, this increases power through separatrix, Psep, and thus can produce excessive heat flux to the divertor which can lead to damage. Due to neutron radiation, the materials are even more susceptible to damage for a given heat flux than in non-reactor devices. One way to protect the divertor is seeding plasma impurities to radiate the energy in the scrape-off layer and in the divertor region before it hits the divertor target plates. The aim of this work is to simulate feedback control of Pfus and Psep, and to find a way to decouple Pfus from Psep. Fusion power is controlled either via plasma density using pellet frequency or via changing the ratio of deuterium and tritium in the injected pellets. To prevent increasing Psep and heat load on the divertor while increasing Pfus a xenon gas can be puffed into the main chamber. However, Psep has to be kept above a threshold to stay in H-mode. Therefore, feedback control of the xenon gas puff into the divertor is modelled. Possible methods of keeping Psep fixed while optimising Pfus are presented here. A model of DEMO has been implemented in ASTRA. ASTRA is a transport code for fusion devices equipped with equilibrium, transport, fuelling, heating and current drive modules. To model the combination with feedback control loops, ASTRA was coupled with Simulink. Simulink is a powerful tool to model and to simulate different dynamic systems. It allows fast and simple development of controllers using its built-in blocks. Therefore, coupling of ASTRA with Simulink gives the advantage of fast development of controllers for the power plant modelled with sophisticated physics based on the transport codes.
A project on the scale of DEMO requires a formal systems engineering approach. Mapping the interfaces, dependencies and relationships between subsystems permits an understanding of a conceptual design from a set of complementary and consistent perspectives. It also helps to prevent clashes and incompatibility between subsystems at a later stage of engineering design. The first stage of this work has focussed on the DEMO Plasma Operation State (POS), where the tokamak executes a pulse sequence. For each of the substates within the POS, information gained from experience on JET has been used to create a matrix detailing which subsystems will be active. This involves attempting to define, characterise and, if possible, quantify the salient attributes and functions for each required subsystem. The resulting information has been incorporated into a model using the systems engineering language SysML. The resulting model, which forms part of a broader Model Based System Engineering (MBSE) activity within DEMO, will be presented. Such a model should provide a framework for analytical decision making as the project progresses and should detail the following perspectives:

A tokamak has composition, comprising a number of subsystems, e.g. the Gas Injection subsystem. The tokamak has one associated Tokamak State Machine, comprising a number of states, and a number of transitions that allow movement from one state to another. Each subsystem has allocated to it a set of operational modes. These subsystem modes provide a behavioural definition. Each subsystem has key parameters. Parameter values in a given state identify the relevant subsystem characteristics of that state. Each tokamak state can involve one or more subsystem modes. In addition, subsystem modes may be involved when in one or more tokamak states. The resultant SysML model provides a mechanism for analysing the inherent complexity of tokamak subsystem modes during specific tokamak states.
Recent DEMO physics study has focused on several issues raised from the JA Model 2014 concept. The concept is characterized by a fusion power of ~1.5 GW and a major radius of 8.5 m based on the technical assessments of divertor heat removal capability, overall tritium breeding ratio TBR > 1.05, full inductive ramp-up of plasma current, and so on. A problem is compatibility between divertor detachment and operational density due to the low Greenwald density limit. Increase in a plasma elongation and decrease in the major radius are essential for increase of the Greenwald density limit. Regarding the plasma elongation, the effect of conducting shells on vertical stability has been investigated by considering the actual structures of in-vessel components and vacuum vessel compatible with maintenance scheme. The 3D eddy-current analysis indicates that a double-loop type shell contribute to improve the plasma elongation from 1.65 to 1.72. Regarding the major radius, the effect of ECH on the saving of CS flux consumption has been investigated. The result indicates that the CS-flux saving of ~30 Wb by the EC power of ~30 MW can contribute to reduce the major radius to ~8.25 m. Both results of higher plasma elongation and smaller major radius can increase the Greenwald density limit by over 10%. It should be noted that increasing the plasma elongation significantly reduces a requirement of energy confinement enhancement factor on H-mode scaling from 1.31 to 1.19. Furthermore, divertor plasma simulation at low density of $1.8 \times 10^{19} \text{m}^{-3}$ at the separatrix shows that full detachment at the inner divertor and partial detachment at the outer divertor are produced with high radiation fraction of 0.7-0.8 by Ar impurity seeding.
P1.042

Conceptual design study of pellet fuelling system for DEMO

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Controllability of output power is one of the essential requirements for DEMO. Fuel control is expected as primary knob for the fusion power control. Pellet injection is considered as primary fueling technique in DEMO as with the ITER. Difference of requirement for fueling system in DEMO compared to ITER comes from demand of larger output. It consequences requirement of more fueling efficiency to obtain higher fuel density as well as sufficient purity under the conditions of density limit against to the larger He ash generation. The fueling efficiency depends on pellet deposition profile. Because of high edge temperature, pellet ablation beyond the pedestal top is hopeless in DEMO. According to an output control simulation study with 1.5D transport code with pedestal suggests the fuel deposition profile peak preferably reaches at least $r/a \sim 0.85$. Thus, the plasmoid drift must be utilized to achieve the expected deposition depth. Because the pellet cloud drifts down the gradient of toroidal magnetic field in the positive R-direction, the Z-coordinate of injected point must locate in-between upper and lower edge of targeted flux tube. In order to translate such requirements of the fuel deposition into engineering requirement specification of pellet injection system, we have investigated injection angle, speed and mass dependence of pellet deposition using a pellet ablation-drift code against to the profiles with pedestal. As the result of scan survey, pellet speed $\sim 2000$ m/s and poloidal angle $> 120$ degrees at plasma surface are suggested as a rough estimation for DEMO R&D target. Since the pellet speed is limited by curvature of pellet guide tube (PGT), reconciliation among the PGT route with small curvature, location of coils and demand of neutron shielding arises as an engineering challenge. Optimization study of PGT route design and impact on neutron fluence on vacuum vessel will be presented.
P1.043

Detection of Neoclassical Tearing Modes in DEMO using the Electron Cyclotron Emission

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Tokamak plasmas, in low safety factor scenarios, are prone to magnetohydrodynamic (MHD) low m,n instabilities which may affect the energy and particle confinement time and possibly lead to disruptive plasma termination. In presently operating tokamaks high space resolution (~2cm) and high time resolution (0.01-0.1ms) Electron Cyclotron Emission (ECE) diagnostics are embedded in the control loop finalized to MHD control, often in synergy with pick-up coils sensitive to the magnetic fluctuations. Microwave diagnostics have plasma-facing components that are electrically passive, have metal body and are mechanically fixed. Such characteristics of robustness and reliability are promising features but it has not been proved yet that the ECE diagnostic performances are good enough for this task in the DEMO reactor. Moreover, the same kind of solution used today plant might be beyond reach in a fusion power, given the much higher neutron fluence (15-20 times of ITER) which makes unlikely the regular operation of detectors close to the vessel wall like pick-up coils. One specific task that the ECE diagnostics should accomplish in DEMO is then the prompt detection of Neoclassical Tearing Modes without the auxiliary detection capabilities of fast magnetic diagnostics. An assessment of this capability can be performed simulating the ECE temperature signals [1] associated with NTM perturbation [2] and then processing them with a detection algorithm [3] without using any other diagnostic signal, also taking into account noise sources. The results of such assessment referred to the EU-DEMO1-2015 scenario is reported in this paper, showing that extraordinary mode ECE in 2nd harmonics seems to have enough space resolution in the region interested by 3/2 and 2/1 NTMs. [1] D. Farina et al, AIP Conference Proceedings 988 128 [2] H. van den Brand et al, Nuclear Fusion 53 013005 [3] J. Berrino et al., Nucl. Fusion 45 1350
P1.044

A 3D Electromagnetic Model of the Iron Core in JET

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The Magnet and Power Supplies system in JET includes a ferromagnetic core able to increase the transformer effect by improving the magnetic coupling with the plasma. The iron configuration is based on an inner cylindrical core and eight returning limbs; the ferromagnetic circuit is designed in such a way that the inner column saturates during standard operations [1]. The modelling of the magnetic circuit is a critical issue because of its impact on several applications, including equilibrium and reconstruction analysis required for control applications. The most used model in present applications is based on Equivalent Currents (ECs) placed on the iron boundary together with additional specific constraints, in 2D axisymmetric frame. The (circular) ECs are chosen, by using the available magnetic measurements, to best represent the magnetic polarization effect [1]. Due to the axisymmetric assumption such approach is not well suited to deal with significant 3D effects, e.g. arising in operations with Error Field Correction Coils (EFCC). In this paper a new methodology is proposed, based on a set of 3D-shaped ECs and able to better model the actual 3D magnetization. According to a well assessed approach [2], the 3D shape of ECs is represented by a set of small filamentary sources. The methodology has been successfully validated in a number of JET experiments where 3D effects are generated by EFCC currents. The new procedure has been designed to be easily coupled with equilibrium or reconstruction codes such as EFIT/V3FIT. [1] O’Brien D. P., et al. “Equilibrium analysis of iron core tokamaks using a full domain method.” Nuclear fusion 32.8 (1992):1351. [2] Chiariello A. G., et al. “Effectiveness in 3-D Magnetic Field Evaluation of Complex Magnets.” IEEE Trans.Magn., vol.51.3, 2015.
Real time control developments at JET in preparation for deuterium-tritium operation

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Robust high performance plasma scenarios are being developed to exploit the unique capability of JET to operate with Tritium and Deuterium. In this context, real time control schemes are used to guide the plasma into the desired state and maintain it there. Other real time schemes detect undesirable behaviour and trigger appropriate actions to assure the best experimental results without unnecessary use of the limited neutron production and Tritium budget. Rather than discussing the traditional plasma position, current and density control systems, this paper will concentrate on more advanced control schemes which are in use or under development at JET. Such controllers fall naturally into two categories: 1: Continuously active controllers, exemplified by the control of Beta via NBI as developed in the 1990s, use various diagnostics together with real time processing to generate actuator requests aimed at maintaining the plasma in the desired state. More recent successful advances include: (i) Control of the degree of plasma detachment via impurity injection; (ii) ELM frequency control via gas/Pellet injection; (iii) Sawtooth pacing using ICRH modulation. 2: Event/threshold detection algorithms triggering a variety of actions: (i) Execute the safest possible termination strategy if the plasma is heading towards a disruption. Different actions, such as reduction of plasma current or the triggering of massive gas injection for disruption mitigation, are executed depending on whether the disruption is imminent or not. (ii) Terminate pulses early, when they do not seem likely to produce the desired plasma conditions. This would save neutron and tritium budget allowing an increased number of successful discharges to be run. Before going into Deuterium-Tritium operation these control schemes should be integrated into the plasma scenarios, assuring that the various controllers are mutually compatible. Work to assure this integration will be of high priority for JET in the next experimental campaign.
Plasma equilibrium based on RF-driven current profile without assuming nested magnetic surfaces on QUEST

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In the present RF-driven (ECCD) steady-state plasma on QUEST (Bt = 0.25 T, R = 0.68 m, a = 0.40 m), plasma current seems to flow in the open magnetic surface outside of the closed magnetic surface in the low-field region according to plasma current fitting (PCF) method. The current in the open magnetic surface seems due to orbit-driven current by high-energy particles in RF-driven plasma. So based on the analysis of current density profile based on the orbit-driven current, plasma equilibrium is to be calculated. High energy particles guiding center orbits are calculated as a contour plot of conserved variable in Hamiltonian formulation and particles initial position with different levels of energy and pitch angles, that satisfy resonance condition, are considered. From collisionless approximation, distribution function is assumed uniform on the particle orbit. Trapped particles, which do not interact with the first wall, contribute to the distribution function (precession current). Then the profile of orbit-driven current is estimated by multiplying the particle density on the resonance surface and the velocity on the orbits. Negative current near the magnetic axis is shown and hollow current profile is expected even if pressure driven current is considered. Considering the hollow current profile shifted toward the low-field region, the equilibrium is fitted within nested magnetic surfaces by J-EFIT coded by MATLAB. Though the plasma boundary shape reflects the plasma current density profile, the tendency of the equilibrium shape fitted by the J-EFIT did not coincide with the orbit-driven current profile. The collision effect on the current profile may be important. But the extension to the current profile without assuming nested contours is introduced into the J-EFIT code and the appropriate plasma shape of free boundary with the hollow current profile may be fitted to the measured magnetic data.
Merging compression startup, pioneered on START, is a successful and robust method for plasma breakdown and plasma current startup which does not involve a solenoid. Tokamak Energy is currently constructing a relatively small (R=0.4m) high toroidal field (BT>2T) spherical tokamak (aspect ratio ~ 1.8) called ST40 which will have ~2MA of plasma current. A consequence of the ambitiously high toroidal field is that the maximum current in the solenoid is limited to ensure that the twisting force on the toroidal field coil assembly is manageable. Consequently, solenoid startup will not be sufficient to produce ~2MA of plasma current. The intention therefore is to use merging compression for breakdown and to produce the operating plasma current. Merging compression on START and MAST produced a maximum plasma current of 155kA and 470kA. Currently, a complete theoretical model for the merging process does not exist. We therefore rely on an experimentally derived scaling law, created using START and MAST data, to extrapolate the required in vessel PF coil current required to produce ~2MA of plasma current (note: ST40’s in vessel PF coils are equivalent to MAST’s P3 coils; about which plasma rings form prior to merging). Using a free boundary MHD equilibrium solver we modelled the plasma as a series of snapshots in time; before and after merging. Unexpectedly we found that eddy currents in the vessel play an important role in the plasma equilibrium. Neglecting eddy currents we find that the equilibrium is extremely sensitive to the vertical field produced by PF coils outside the vessel, which is in contradiction to the experience on START and MAST which showed merging compression to be an extremely robust and repeatable technique. We therefore conclude that it is essential to take vessel eddy currents into account when modelling merging compression in tokamaks.
The main objective of this work is to demonstrate that a digital integrator based on the chopper modulation concept is capable of meeting the ITER requirements. The ITER magnetics diagnostic requires a maximum drift of 500 uV.s/hour, among other specifications, for the respective signal integrators. As of today, known COTS integrator modules do not fully comply simultaneously with all ITER requirements. A chopper design, implemented by IPFN based on a W7-X concept, was used as a starting point for the presented developments. The suitability of this integrator module for use on ITER was previously tested by IPFN and several issues were identified. New designs were subsequently developed in order to correct these problems or identify their causes. A first phase of prototyping, presented in this work, comprises the development and testing of 4 resulting design variants. Combinations of a SAR ADC (AD7960) and a Sigma-Delta ADC (ADS1675) with different analog front ends were used for the corresponding integrator prototypes. The designs have a common interface to an FPGA based system, developed by CCFE, that receives the acquired data and sends it through GbE to a PC for processing the signal integration. The GbE network acts as the interfacing medium for the tests, allowing connection of the integrator prototype to a permanent data storage MDSplus environment. The integrator prototype designs and tests done so far will be presented. This work was funded by F4E via contracts F4E-OPE-442 and F4E-OPE_361-08. The views expressed herein are the sole responsibility of the authors and don’t necessarily reflect the views of F4E or the ITER Organization. Neither F4E nor any person acting on behalf of F4E is responsible for the use, which might be made, of the information contained herein. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.
Design development, integration and assembly of the ITER steady-state magnetic sensors

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The final design of the steady-state sensor diagnostic, developed collaboratively by ITER Organization and IPP Prague, is presented. The steady-state sensors – a subsystem of the ITER magnetic diagnostics – will contribute to the measurement of the plasma current, plasma-wall clearance, and local perturbations of the magnetic flux surfaces near the wall. The diagnostic consists of an array of sixty sensors mounted on the vacuum vessel outer shell and distributed approximately uniformly in the poloidal plane. The match-box size sensor housing accommodates a pair of bismuth Hall sensors [1] with the measurement axes parallel (in the poloidal plane) and normal to the vacuum vessel, and a thermocouple to compensate the variation of the Hall sensor output with the temperature. The housing also comprises a triplet of reflector nests for the as-installed metrology. The small mass/size of the housing helps to reduce the forces during a seismic event and due to Halo currents, and lowers the risk that the sensors won’t fit in the as-fabricated gap between the vacuum vessel and the thermal shield. Small-size housing also allows for a single-point attachment to the vacuum vessel which eliminates stresses on the housing due to thermal expansion and the loop force. The sensor attachment is designed to fulfil the stringent criteria for the weldments to the ITER vacuum vessel, and to allow for automated welding and the inspection of the root of the weld. The attachment features an isthmus-type stress relieve joint which is also to prevent the damage of the Hall sensors during the welding process, as was verified in a dedicated weld experiment and in thermal simulations. The contribution will also address the prototype R&D tests and magnetic simulations as well as the project schedule up to the sensors delivery in 2019. [1] I. Duran et al., this conference.
Development of Bismuth Hall sensors for ITER steady state magnetic diagnostics

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Hall sensors with their small dimensions, simple principle of operation, and large dynamic range offer an attractive non-inductive method of magnetic field measurements for future fusion reactors operating in steady state regime. The applicability of commercially available Hall sensors, which are based on semiconductor sensing layer, is strongly limited by insufficient range of operational temperatures and limited radiation hardness. Hall sensors with metallic sensing layer offer interesting alternative compared to the semiconductor devices. Due to their very low sensitivity, the metal-based sensors are practically omitted in both commercial and research spheres. On the other hand, their expected advantages such as higher radiation hardness and high temperature resistance can possibly prevail over this weakness in case of their application in future fusion based power generating systems. Recently, the Hall sensors based on bismuth sensitive layer were selected to be implemented on ITER as ex-vessel steady state magnetic sensors. The proposed contribution will review the present optimized design of these sensors and their manufacturing technology, including some alternative options. The sensor prototypes were extensively tested to evaluate their compatibility with ITER requirements. Characterisation of the sensors properties was done using AC detection technique to ensure high noise immunity. The measured quantities include: offset voltage, sensitivity and its dependence on temperature, input and output resistance, linearity, charge carrier density and mobility, as well as performance of the sensor after temperature cycling and neutron irradiation up to the $10^{19}$ cm$^{-2}$.2.
Signal processing for the extreme environment Hall sensors

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A prototype electronics for the ITER ex-vessel steady state magnetic field metallic Hall sensors based on the analog lock-in signal processing with dynamic quadrature offset cancelation was developed and tested. Testing was carried out on Bismuth Hall sensors placed in the SAMM test assembly. The magnetic coils are used for measuring the magnetic field of the fusion reactor conventionally. However, the inductive sensors measure only the magnetic field changes. The ITER and DEMO applications need the steady-state magnetic field sensors like Hall sensors. Commercially available Hall sensors based on semiconductors have a very limited range of operating temperatures and sensitivity to radiation. Metallic Hall sensors provide temperature and radiation resistance but are much less sensitive. Therefore, the output signal of the metallic Hall sensors has to be processed by special methods allowing measurement of the signal microvolt level with the high noise and distortion immunity. The paper describes signal processing methods, prototype electronics, and its testing. Three measuring methods are evaluated – DC signal processing, AC digital lock-in signal processing and AC analog lock-in signal processing with dynamic quadrature offset cancelation. The accuracy and linearity of the methods are compared in a magnetic field range of 0 – 0.5 T.
The Plasma Position Reflectometry (PPR) diagnostic will be used in ITER to measure the plasma position/shape in order to provide a reference for the magnetic diagnostics during very long (>1000s) pulse operation, where the position deduced from the magnetics is known to be subject to substantial error. It consists of five reflectometers distributed at four locations, known as gaps 3-6, operating in O-mode in the frequency range 15-75 GHz. The systems of gaps 4 and 6, which are considered here, are known as the PPR in-vessel systems, since its bi-static antenna system and feeding waveguides are installed inside the ITER vacuum vessel – for gap 4 the antennas are on the low-field side, for gap 6 on the high-field side. A critical issue in the design of these systems is the transmission line (TL) to/from the antennas since it uses oversized rectangular waveguides that, being welded to the vessel inner-shell must conform to an intricate path/geometry. This includes a 90° bend right behind the antennas, for both gaps 4 and 6, and a 120° bend just before entering the port extension, exclusively for gap 4. However, oversized bends can excite higher order modes and create resonances, which increase the transmission losses and could significantly affect the diagnostic’s ability to meet the measurement requirements. Hence, careful assessment of these components will be crucial to the diagnostic’s success. Here, the performance of the 90° and 120° bends is studied via numeric simulations and, in the case of the 90° bend, compared to laboratory measurements of a prototype which revealed an excellent performance, with overall losses around 0.5 dB and no resonances across the whole frequency range.
Electromagnetic analysis of the in-vessel ITER plasma-position reflectometry antennas

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ITER Plasma Position Reflectometry (PPR) system will be used to estimate the distance between the position of the magnetic separatrix and the first-wall at four pre-defined locations also known as gaps 3, 4, 5, and 6, complementing the information provided by magnetic diagnostics. For gaps 4 and 6, the antennas are to be installed in-vessel between two blanket shield modules. The microwave signal is routed to/from the antennas using rectangular oversized waveguides that enter/exit the vacuum vessel (VV) through feed-outs located in upper ports 01 and 14, respectively. Being in-vessel and attached to inner shell of the VV, these components will be subject to high mechanical loads during plasma disruptions. The magnetic field gradients associated with these disruptions induce eddy and halo currents in the surrounding metallic structures that interact with the remaining magnetic field producing forces and torques that may compromise the integrity of the components. Although the antenna assembly is designed in such way that it can be remotely installed/removed, the waveguides are installed in a support structure welded to the inner shell of the VV and no maintenance is foreseen during ITER operation. Therefore, the design of these components requires detailed integrity analysis to assess if the components will be able to remain operational and fulfill their function for the lifetime of ITER. Here, we report on preliminary electromagnetic analysis of the in-vessel components of gap 4 system, including the development of finite element models of the VV and PPR system components. The analysis was performed with ANSYS based on inputs from DINA code obtained from F4E/IO, to assess the electromagnetic loads on these components for different plasma disruption scenarios. The loads obtained through the analysis presented here will be used as input in the global integrity analysis of the in-vessel components of the ITER PPR system.
Diamond Window Diagnostics Concepts For Fusion Reactors - Updates of the Design

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The future nuclear fusion power plants will require Electron Cyclotron Heating and Current Drive (ECH&CD) systems to heat up and stabilize the plasma inside the vacuum vessel. One of the key components of such systems is the Chemical Vapor Deposition (CVD) diamond window. The purpose of this device is to act as vacuum and tritium boundary while providing a high microwave transparency with minimal reflectivity. Although suited for high power microwave operation, the windows shall be internally monitored in order to properly ensure the ECH system efficiency and safety. In this paper, the latest assessment study on a set of diagnostics to be part of the window assembly is shown. The required diagnostics include arc and tritium detection, microwave stray radiation (perpendicular to the main beam and generated by cracks in the windows), pressure and disk temperature measurements. The devices must have a compact, simple and flexible layout, with a rugged design, to maximize serviceability and durability. When multiple options are possible for the some of the diagnostic systems (e.g., scintillation devices vs solid state detectors for tritium detection), tradeoffs were assessed. To accommodate the diagnostics previously mentioned, a new design for the window housing was developed. As the design of the original diamond window assembly underwent further development since the beginning of this project, an update of the general layout was required. The new layout presented here integrates the updates of the windows assembly with those to the diagnostics. To validate the concepts, a test bench was developed to carry out measurements under conditions similar to the operative ones.
Conceptual design and dynamic simulation of a fast-ion loss detector for ITER

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Scintillator based fast-ion loss detectors (FILD) are used in virtually all major tokamaks and stellarators to study the fast-ion losses induced by magnetohydrodynamic (MHD) fluctuations. FILD systems provide velocity-space measurements of fast-ion losses with alfvénic temporal resolution. This information is crucial to identify the MHD fluctuations responsible for the actual fast-ion losses and to understand the wave-particle interaction underlying the transport mechanism. The ITPA Topical Group on Energetic Particles has selected a FILD as the most preferred diagnostic for fast-ion loss measurements in ITER. In response to this prioritization, the Port Plugs and Diagnostics Integration Division at ITER Organization (IO) initiated an effort to develop a conceptual design of a reciprocating FILD in ITER. The extreme working conditions expected in ITER impose especial and unique requirements for such a system. A fast and reliable motion of the detector head, with approx. 10 cm diameter and 20 cm stroke, is mandatory to obtain meaningful measurements of fusion born alpha particle losses with acceptable thermal loads. The dynamic system has been designed as to avoid disruption halo currents. This fast motion will be controlled by an energized solenoid which will create the needed torque, taking advantage of the tokamak magnetic field. In this contribution, a conceptual mechanical design and a dynamic simulator for the ITER FILD are presented. This simulator models the FILD mechanical behavior as a multi-body system real-time controlled by a proportional-integral-derivative (PID) algorithm. The PID sets the voltage applied to the solenoid depending on the actual and target position in real time. Aspects such as friction in joints are taken into account in the model, allowing to determine reaction forces under high friction conditions, as those related to in-vacuum environments. Simulation results describing the detector dynamic performance and mechanical strength under several working scenarios will be presented.
Conceptual studies for the management of thermal properties of ITER bolometer cameras

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As part of ITER’s fusion diagnostic systems, metal foil – miniaturised metal resistor type bolometer cameras are envisaged to provide the measurement of the total plasma radiation. For this kind of bolometer sensor the temperature of a measurement and a reference absorber is realised by metallic meanders on their back side, which are combined in an electrical configuration of a Wheatstone bridge. For the reference absorber being able to reliably compensate changes in the environmental temperature, it is necessary to assure that the temperature gradient within one bolometer channel, consisting of a measuring and reference meander, is as low as possible, preferably <0.1°C. In order to successfully establish this condition, the entire design of the sensor mount, camera housing and wall attachment have to be considered. Due to the high number of differently oriented cameras required in ITER, the main goal of this study is to map the thermal behaviour of these devices under typical loads and, identifying the underlying principles to provide a universal guidelines and starting points for future designs, in order to prevent one by one optimization of each separate camera. Starting with the design presented at the conceptual design review, features affecting the heat flow path were identified and parametrised for the thermal finite-element model. Simulation results show that after simultaneously optimizing a set of selected geometrical parameters for each case, the current construction is able to produce the desired thermal values and gradients for every investigated camera orientation.
Real-time calibration parameter compensation of metal resistive bolometers operating in a thermal varying environment

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The ITER bolometer diagnostic will have to provide accurate measurements of the plasma radiation in a varying thermal environment of up to 250°C. Current fusion experiments perform regular in-situ calibration of the detector properties, assuming stable calibration parameters within short discharge times, e.g. 10 s on ASDEX Upgrade. For long-pulse fusion experiments, e.g. W7-X, the diagnostic is operated with water cooling for achieving a stable temperature environment. However, ITER will be equipped with over a hundred bolometer cameras and is planned to have discharge times of up to 1 h. Due to space restrictions, active cooling is not available for all locations. Thus, an alternative approach is required to allow for compensation of the changing calibration values due to thermal drifts. This paper demonstrates a method using the Wheatstone-bridge current of the detector to calculate in real-time the changing calibration values, such as the heat capacity, the thermal time constant and the meander resistances. It is shown, that the thermal offset error, a calibration parameter drift associated with the production tolerances between measurement and reference meander resistances, can be calculated by extrapolating from the initial spread of the resistances. Measurements in the ITER Bolometer Vacuum test facility (IBOVAC), used to simulate ITER-relevant thermal and vacuum environment, show that the change of the calibration values can be predicted during repeated thermal cycles over durations sufficient for ITER discharges and even longer. Confidence intervals for each parameter of the in-situ calibration method are determined and compared with the accuracy of the proposed extrapolation method for ITER showing that this method provides an equivalent quality of the measurement results.
In ITER, like in any fusion reactor, the plasma-wall interaction is unavoidable. It leads to material erosion and potential re-deposition or other surface morphology changes, as well as dust formation and tritium retention. The decision to start ITER operations with a full-W divertor has significantly reduced the expected erosion of the divertor target making observation of the target during discharges unnecessary. Co-deposition of beryllium is expected to be limited in the high heat flux region near the strike-point because of the high surface temperature in this area. Strong surface morphology changes can, however, happen in the strike-point area due to roughening/cracking and possibly melting caused by ELMs. A diagnostic for surface morphology measurements between discharges is deemed necessary to check for the appearance and development of tungsten surface damage, initially in the micrometer range. To monitor the fine morphology changes of the target surfaces non-invasive remote sensing method is preferred, and after a dedicated workshop on Erosion, Deposition, Dust and Tritium retention, the dual beam speckle interferometry method has been recommended for this diagnostic. This technique based on the Michelson interferometer scheme is proposed for measuring the surface topology of the vertical parts of the inner and outer divertor targets, where plasma – wall interactions are the most intense. The evolution of the morphology will be characterized based on observations made before and after plasma shots. To allow a proper analysis (absolute calibration), reference surfaces are needed. They are currently placed ~35mm behind the surface of the tungsten monoblocks. The proposed diagnostic is unique in its concept and realization and has so far not been tested in a real working tokamak environment. This paper will review the requirements and objectives of this diagnostic, present the opto-mechanical design and detail the numerous challenges faced (limited space, high loads, vibration sensitivity, etc).
The preparation of the Shutdown Dose Rate experiment for the next JET Deuterium-Tritium campaign

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The assessment of the Shutdown Dose Rate (SDR) due to neutron activation is a major safety issue for fusion devices and in the last decade several benchmark experiments have been conducted at JET during Deuterium-Deuterium shutdown for the validation of the numerical tools used in ITER nuclear analyses. The future Deuterium-Tritium campaign at JET (DTE-2) will provide a unique opportunity to validate the codes under ITER-relevant conditions and a novel SDR experiment is in preparation in the frame of the NEXP-JET3 subproject within EUOfusion Consortium. The experimental setup for the future SDR experiment has been accurately designed to reduce risks and uncertainties and preliminary tests and calibration of the active system were performed. Spherical air-vented ionization chambers (ICs) will be used for on-line ex-vessel decay gamma dose measurements during JET shutdown following DT operations. Activation foils have been selected for measuring the neutron fluence near the ICs during operations. The systems will be located at the side port of Octant 1 (close to the radial neutron camera) and on top of the ITER-like Antenna (ILA) in Octant 2 on proper low activation shelves. Neutron irradiation tests were carried-out at the Frascati Neutron Generator (FNG) with the aim of assessing the correct functioning of the detector after 14 MeV neutron irradiation and for checking the self-activation of the detector induced by neutrons. The present work is dedicated to the preparation of the Shutdown Dose Rate experiment for the next Deuterium-Tritium campaign. The selected experimental equipment together with results of calibration and irradiation tests of ICs are described. Tests at the FNG confirmed the capability of the dosimetry systems to perform on-line decay gamma dose rate measurements, to follow gamma dose decay at the end of neutron irradiation as well as insignificant activation of the detectors.
P1.061

Hardware architecture the JET Neutron Camera Upgrade (NCU) new data acquisition and processing system

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The Neutron Camera is a Joint European Torus (JET) diagnostic with the main function of measuring the 2.5 MeV (DD) and 14 MeV (DT) neutron emissivity profile over a poloidal plasma cross-section using line-integrated measurements along a number of collimated channels (lines-of-sight, LOS). Measurements are performed using two detectors: NE213 liquid scintillators (DD, low power DT) and BC418 plastic scintillators (high power DT, low intensity DT in strong DD). The BC418 detectors work using old analog electronics, suffering limitations: No raw data storage. No accurate pulse height spectra (PHS). Complex calibration procedure. Moreover, as during the 1997 DTE1 campaign the BC418 detectors worked up to \( \sim 1.2 \times 10^6 \) cps (above 10 MeV threshold), higher NBI power in the future DT campaign might imply even higher rates. The NE213 detectors are coupled to a Field Programmable Gated Array (FPGA)-based digital system. This overcomes the BC418 limitations, but off-line processing is limited to \( \sim 9 \times 10^5 \) cps (lab. tests), with limited pile-up management. To address these limitations, in view of JET DT campaigns, an enhancement project (Neutron Camera Upgrade, NCU) was launched with two main objectives: i) to increase the performance and reliability of the 14 MeV neutron measurements performed by BC418 detectors; ii) to assess the possibility of increasing the counting rate capabilities of the NC detection system based on NE213 detectors. The first objective will be achieved by installing at JET a new FPGA-based digital system. The new units will include:

- High throughput digital acquisition for BC418 detectors performing on-line preprocessing.
- Raw data storage, in the NC cubicles.
- Off-line processing (pile-up, DT neutron count rates, PHS, calibration).

The present paper describes the hardware architecture and the FPGA processing selected for the new NCU system and the tests carried out at JET for its design.
Calculation of the profile dependent neutron backscatter matrix for the JET neutron camera system

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The signal of a neutron detector can be divided into an unscattered and a scattered component. In fusion, the unscattered, direct component reaches the detector directly from the fusion plasma. The scattered neutrons, on the other hand, reach the detector after interacting with some of the materials in the fusion device. More specifically, the backscatter component is defined as the signal from neutrons that are scattered in the wall of the tokamak directly opposite to the neutron detector. Backscattered neutrons can contribute significantly to the total neutron rate seen by a neutron detector and it is therefore important to accurately estimate their energy distribution and rate. Previously the calculation of the backscatter component was done by first estimating the expected direct emission and then multiplying it with an energy dependent neutron backscatter matrix. The latter was obtained by combining many MCNP simulations of the backscattered neutron energy distribution for monoenergetic neutron emission, each simulation with a different energy. However this method neglects the fact that the backscatter component can depend on the neutron emissivity profile. Here we take profile effects into account by producing a neutron backscatter matrix that is profile dependent (instead of energy dependent). This is done by dividing the plasma source into toroidal voxels in the MCNP simulation and constructing the matrix from the backscattered neutron energy distribution from each voxel. The backscatter component is then obtained by multiplying the emissivity profile with the matrix. We apply this method to calculate the neutron backscatter matrix for the neutron camera detectors at JET. We then use the matrix to evaluate the backscatter component for different neutron emissivity profiles, showing that they produce different results. We conclude that this method improves the evaluation of the neutron backscatter component and should be used for future analyses of the neutron emission.
Characterization of a neutron generator for the JET monitoring system calibration with NE-213 spectrometer

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The second experimental deuterium-tritium (DT2) campaign is planned at JET in 2019. Calibration of the JET neutron emission monitoring system, consisting of fission chambers (KN1) and an activation system (KN2), will be carried out with a compact deuterium-tritium neutron generator (NG) with suitable intensity (∼5x10⁸ n/s). The accuracy goal for this calibration is <10% uncertainty at 14 MeV neutron energy. To achieve this goal, the compact NG was intensively examined at the National Physical Laboratory (NPL) Teddington and neutron emission spectra measured at different angles with respect to the NG axis. The examination was done utilizing diamond diodes, long counters, silicon diodes, foil activation techniques, and a NE-213 scintillator spectrometer. In parallel, a MCNP model of the NG is developed which is checked and validated by the results of the neutron spectra measurements. This MCNP model will play an essential role in the JET neutron monitor calibration. Here we report on the measurement of fast neutron spectra emitted from the compact NG recorded with a NE-213 scintillator based neutron spectrometer. The NE-213 scintillator was placed at several angles with respect to the NG axis covering a full circle with a radius of 146 cm. The measurement provides data for comparison with the MCNP model of the NG, in particular the position of the DT neutron peak in the neutron spectrum and a possible DD neutron contribution.
Beam emission spectroscopy diagnostic based on neutral beam in EAST tokamak


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Abstract Beam Emission Spectroscopy (BES) diagnostic based on neutral beam injection (NBI) has recently been developed in EAST tokamak. A 128-channel Hamamatsu S8550 APD detector array is chosen as the core device. Three cavity interference filter with a center frequency of 659.33nm and a bandwidth of 1.59nm is used to eliminate the interference Dα signal and carbon impurities radiation. This BES system diagnoses the plasma density fluctuation with a sample rate of 1MHz and a spatial resolution of 1-3cm. It can diagnose a rectangular area in one shot, with a radial length of 20cm and a vertical length of 10cm in the plasma cross section, which is movable from the core plasma to the edge of low field side of EAST by means of changing the angle of beam splitting lens. Space calibration and filter test are also presented. Data in the recent experiments of BES show the great change of density fluctuations in the L-H transition.
Divertor heat flux study of L-H transition with different auxiliary heating in EAST

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H-mode is the main operation mode in the future fusion reactor and L-H transition is one of the concerning issue of H-mode research[1]. Much effort has been made on the research of L-H transition, however, the detail characters of the L-H transition need more research to afford reference for the optimization of H-mode plasma discharge [2-4]. An infrared(IR)/visible endoscope system was built on the Experimental Advanced Superconducting Tokamak (EAST) in 2014. The temperature distributions of the lower divertor during L-H transition with the lower-hybrid wave current drive (LHCD) only as well as with both the LHCD and the NBI at different time have been measured. Based on the IR data of EAST, the heat fluxes on the lower outer divertor were calculated with a code named DFLUX developed by ASIPP, aimed to provide reference for the H-mode operation of EAST. The analyzed discharges were lower single null diverted discharges. Analysis results show that the changes of heat flux before and after L-H transition are related to the types of auxiliary heating and the time of energy injection. When the auxiliary power in the case with LHCD only (~2MW) is one-time injection, two even three L-H transitions could occur. The peak heat fluxes increased rapidly with the energy injection and shut down at the time of L-H transition. When the auxiliary power in the case with LHCD(1.2MW) and NBI (~1MW) are injected successively, the plasma density after L-H transition were often step growth and the changes of peak heat fluxes of L-H transition were relatively small.
The WEST Plasma Facing Components Protection

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The WEST (W Environment for Steady-state Tokamak) \cite{1} project consists in a major upgrade of the superconducting medium size tokamak Tore Supra to minimize risks for the ITER divertor procurement in terms of cost, delays and performance. This modification consists in changing the present circular magnetic configuration to a divertor configuration and implementing an ITER like actively cooled Tungsten divertor. Heat load on divertor target will range from a few MW/m\textsuperscript{2} up to 20MW/m\textsuperscript{2} depending on the X point location and the heat flux decay length. To operate this new W actively-cooled facility the Wall Monitoring System (WMS) has been set-up to deal with such issues. The WMS will be based on an intensive use of image and multi-sensor analysis. The WMS has been divided in three parts; before discharge a pre-pulse analysis tool to check compatibility between plasma scenario and PFCs operational limits; during discharge a real-time system taking into account all necessary measurements involved in the PFCs protection; after discharge a set of analysis tools accessing WEST database for plasma wall interaction understanding and comparison between prediction and experimental results. This paper provides a complete description of the WMS architecture with preliminary results for the three WMS parts: the pre-pulse Power Load Analysis Tool (PLAT\textsuperscript{o}) applied to foreseen WEST plasma scenarios on both inertial and actively-cooled components; DWMS real-time system architecture, features and performances and post-pulse analysis software package ThermaVIP capabilities. 1. J. Bucalossi et al., Fusion Engineering and Design 89 (2014) 907–912
Measurements and controls implementation for the WEST project

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The WEST project consists in a major upgrade of the superconducting medium size tokamak Tore Supra to minimize risks for ITER divertor procurement and operation. This modification consists in changing the present circular magnetic configuration to a divertor configuration and implementing an ITER like actively cooled Tungsten divertor. Heat load on divertor target will range from a few MW/m² up to 20MW/m² depending on the X point location and the heat flux decay length. To reach these goals while ensuring the protection of the machine, major changes and significant developments are on-going on the measurement systems (diagnostics); the control, data access and communication (CODAC); the plasma control system (PCS), the monitoring and protection of the first wall and modelling to prepare the restart of the plasma. This paper provides an overview of the diagnostics implemented on WEST addressing mainly wall interaction diagnostiques such as visible spectroscopy and imaging diagnostics. The modification of the CODAC and communications networks consists mainly in the development of new acquisition units based on PXIe chassis and the implementation of a new real time network. The new functionalities and architecture of the WEST PCS are detailed; especially it ensures the orchestration of many subsystems such as diagnostics, actuators and allows handling asynchronous off-normal events during the plasma discharge. In correlation, the plasma discharge is now seen as a set of elementary pieces (called segments) joint together. Development of new plasma controllers: plasma shape, position and density control has been performed using control oriented modelling and simulations. Finally a specific activity devoted to ensure the wall protection will be discussed. It consists of pre-pulse modelling of heat flux deposited onto the first wall to raise warning before running the discharge, real time analysis and protection and post pulse analysis.
Measurement of surface temperature of the plasma facing component with Multi-Spectral Infrared thermography diagnostics

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For the long-pulse high-confinement discharges in future tokamaks, the equilibrium of plasma requires an interaction and energy exchange with the first wall materials. The heat flux resulting from this interaction is of the order of 10 MW/m² for steady state conditions and up to 20 MW/m² for transient phases. As a result, surface temperature measurement of the plasma facing components (PFCs) up to more than 3000°C is a major concern to ensure safe operation of large fusion facilities. In tokamaks, infrared (IR) thermography systems are routinely used to monitor the surface temperature of the PFCs. This measurement requires an accurate knowledge of the surface emissivity. To solve this problem, a multi-spectral infrared measurement is proposed as a promising solution. The system has the advantage to carry out a non-intrusive measurement on thermal radiation whilst evaluating surface temperature without requiring a mandatory surface emissivity measurement. In this paper, a conceptual design for the multi-spectral infrared thermography is proposed for detection wavelengths range from 1.5 to 5 mm. The numerical study of the multi-channel system based on the Levenberg-Marquardt (LM) nonlinear curve fitting is applied. The optimization for system wavelength choice is presented. The numerical results presented in this paper demonstrate that this method allows for measurements up to 3000°C with a relative bias of 10%. Furthermore, laboratory experiments have been performed from 200°C to 740°C to confirm the feasibility for temperature measurements on stainless steel and tungsten with emissivity variation from 0.1 to 0.4. In the experiment, most of the unfolding results from the multi-channel detection provide a relative bias of 5% below 740°C, which agrees with theoretical analysis and demonstrates the feasibility for metallic surface temperature measurement with this technology.
Conceptual design of laser transfer system of the JT-60SA Thomson scattering diagnostic

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JT-60SA Thomson scattering system will measure electron temperature and density profile. A YAG laser will be toroidally injected to the JT-60SA on its equatorial plane. If the beam profile changes from flat-top to peaked profile, the laser beam breaks the vacuum window. Thus, we designed beam transfer optics as long as \(~50\) m using a relay image technique. The beam transfer optics designed for the JT-60SA tokamak can transfer the image of initial beam profile (flat-top). The laser beam is transferred from the laser room to the last convex lens placed before the plasma and its size is suppressed within a preferable scale (30 mm). Fused silica, which does not have significant irradiation damage is planned to be employed as materials for the lenses and windows. The resultant beam width in the JT-60SA plasma can be minimized as less than 1 mm. Coping with stray light is another important issue. When the laser goes through the vacuum window, diffuse reflection at the window generates stray light. The stray light can significantly affect signal-to-noise ratio because the Thomson scattered cross section is very small (\(7\times10^{-29}\) m\(^2\)). Numerical ray tracing to simulate suitable number of baffle boards for JT-60SA suggested that more than four baffle boards are necessary to suppress the stray light. We adapted six baffle boards which reduces stray light by \(~10\%\) compared with the case without baffle boards.
Progress of the magnetic sensors development for JT-60SA

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JT-60SA, which has fully super conducting coils, is designed and now being constructed for demonstrate and develop steady-state high beta operation in order to supplement ITER toward DEMO. In order to obtain the information for the control and the physics research on JT-60SA plasma, we developed the many types of magnetic sensors. Compared to JT-60U, JT-60SA needs larger magnetic sensors and longer cable due to its bigger machine size and long port. Therefore, we have designed the magnetic sensors for JT-60SA with taking installation and maintenance into account. In order for easy installation, the connector between a sensor and mineral insulation cable (MIC) and connection box with neither welding nor brazing have been developed. When disruption occurs, a high voltage is applied to the Loop of the magnetic sensors due to change of magnetic flux. On the one hand, massive gas leads the disruption. Therefore the sensors including connection boxes have to have a withstand voltage more than 1kV in the intermediate gas pressure. We successfully increase the withstand voltage in the gas pressure region around the Paschen minimum voltage. Moreover, the newly designed sensors, Rogowski coil, diamagnetic loop and Magnetic probes have been developed. We measured frequency characteristics of these sensors. We will report the manufacturing and tests of the magnetic sensors in JT-60SA.
Feasibility study on the JT-60SA tokamak beam emission spectroscopy diagnostic systems

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The JT-60SA superconducting tokamak is proposed to be equipped with a Lithium Beam Emission Spectroscopy (LiBES) and Deuterium Beam Emission Spectroscopy (DBES) diagnostic systems. The purpose of the LiBES system is SOL and plasma edge density profile measurements and density fluctuation measurements in the SOL and outer edge regions, whereas the DBES system on the heating beams would have the capacity of density fluctuation measurements in the edge and the core regions, as well. The determination of the optimal location for the observation system, as well as the beam injection for the LiBES system, is of vital importance, for which the RENATE synthetic BES diagnostic simulator[1] was employed. The RENATE synthetic BES diagnostic calculates the light emission along 3D modelled neutral beams for various beam materials using the collisional radiative model. A 3D observation module[2] returns the detected photon current values for an arbitrary detector array by accounting for arising geometrical effects. Expected spatial resolution is studied in detail, which is defined as the emission smearing caused by a localized density perturbation. Deconstruction of the spatial resolution is possible into three components that can be individually studied and optimized: the contribution of the magnetic geometry, the contribution of the sensitive volume of the detector and the contribution of the atomic physics. Considerations for an optimal LiBES system require sufficient beam penetration beyond the separatrix and reasonable SOL emission with regard to total emission, while an optimal DBES system requires mostly adequate spatial resolution and a sufficient Doppler shift to enable discrimination of beam emission from emission of the background plasma. Various concepts and geometries of the LiBES and DBES systems for JT60-SA are analysed in detail and several alternative arrangements are put forward for consideration. [1] I.Pusztai etal. 2009 RSI 80 083-502 [2] D.Guszajnov etal. 2012 RSI 83 113-501
In order to extend the operational space of RFX-mod in both RFP and Tokamak configurations, a major refurbishment of the load assembly is under study. It includes the removal of the vacuum vessel to increase the plasma-shell proximity and modifications of the support structure to obtain a new vacuum-tight chamber. This entails the design of a new electromagnetic measure system, taking into account the equilibrium and MHD control system requirements in both RFP and shaped Tokamak configurations. The required bandwidth is 0-500 Hz for the control system and up to 50 kHz for the diagnostics of the high m-n tearing mode rotation. The presence of the copper shell forces to mount most of the sensors onto the inner surface of the shell itself, in vacuum, protected by the first wall graphite tiles. Triaxial pick-up sensors are being considered for both the reduced available room and their convenience for the compensation of alignment errors. The spatial resolution is constrained by the number of tiles: 28 (poloidal) x72 (toroidal) fully covering the toroidal inner surface. A basic set of 7x72 sensors is foreseen to assure the calculation of the MHD mode spectrum up to (m=0,...,3; n=0,...,±35). Moreover, at least 6 poloidal arrays will include 14 sensors to compute the harmonic content needed by the plasma boundary reconstruction algorithm (m=0,...,6). Being available both the magnetic field radial and poloidal component, the poloidal flux loops could be reduced to 2. On the contrary, 12 equally spaced measurements of the toroidal flux, at the middle of each toroidal winding sector, would be of interest for RFP experiments, complemented by the corresponding arrays of triaxial measurements to detect the tokamak diamagnetic component. The installation of an additional set of 4x48 saddle probes is also envisaged to assure a redundant full MHD active control capability.
Development of optical probe for local emission profile measurements in VEST

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Optical emission spectroscopy with inversion process is used to obtain local emission spectrum from line integrated spectra. Tomographic inversion techniques are widely used with complicated noise reduction and sufficient viewing line of sights. On the other hand, optical probe has advantage of direct measurement although it may lead to plasma perturbation. An optical probe with outer diameter of 13 mm is developed and installed in VEST (Versatile Experiment Spherical Torus) to measure local emission spectrum, which can be used for one-dimensional radial profiles of impurity emission intensities and ion temperatures via shot-to-shot measurements at various radial positions. In the optical probe system, collimated light is collected and transmitted to the spectrometer with ICCD (Intensified Charge Coupled Device) via vacuum feed-through. In initial measurements of ohmic plasmas in VEST, radial emission profile of H alpha line (656 nm) shows hollow shape while OV line (650 nm) shows centrally peaked shape. However, plasma current decreases by 10% when the optical probe is inserted up to the plasma center. Modified optical probe is under development to reduce plasma perturbation.
P1.078

Design of combined system of charge exchange spectroscopy and beam emission spectroscopy in VEST

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Helium transport study is essential in burning plasma to prevent fuel dilution from the helium ash accumulation. Charge exchange spectroscopy (CES) is widely used to measure impurity density as well as toroidal rotation and ion temperature. Single-handed CES system have a low accuracy in impurity density measurement due to the large errors in absolute intensity calibration and neutral beam modelling calculation. For this reason, the combined system of charge exchange spectroscopy and beam emission spectroscopy (BES) has been proposed to measure impurity density with only relative intensity calibration of the spectrometer. Doppler Shifted Hα line (rest wavelength at 656.3 nm) for BES and HeII (n=4-3 468.6 nm) for CES are used in the hydrogen discharge with puffing a small amount of helium in VEST. A dichroic beam splitter is used to measure CES and BES signals simultaneously with sharing the same observation and transmission components which include the line of sight, lenses, optical fiber bundles. High spectral resolution spectrometers suitable for the VEST is carefully designed based on the transmission grating with high diffraction efficiency. Multi-chord sightlines in toroidal view have been chosen to have a good spatial resolution from the edge to the core. In this paper, we present a detailed design of the combined system of CES and BES for VEST, with consideration of the estimated spectrum intensities for active signals as well as background noises.
Development of multi-pass Thomson scattering diagnostic system for VEST

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A Thomson scattering(TS) system is developed and commissioned for measuring and analyzing spatial profiles of electron temperature($T_e$) and density($N_e$) of Versatile Experiment Spherical Torus(VEST). Since the estimated $N_e$ of VEST plasma is $\sim 5 \times 10^{18}$m$^{-3}$ which is lower than typical $N_e$ in other tokamaks, each part of the system is carefully designed to maximize the number of collected photons. The system consists of 3 parts such as a novel multi-pass laser injection system, collecting optics, and a polychromator. The multi-pass scheme of a laser injection system including a Faraday rotator and a half-wave plate rotates 90 degrees of the plane of polarization in two roundtrips of a laser beam. The main purpose of this scheme is to achieve a high signal to noise ratio by rejecting the stray light signal or background noise using one laser pulse. The collecting optics comprising a vacuum window and two aspheric lenses is designed for measuring TS signals at the 5 different spatial points with $R=0.4$m as the center of the VEST device and for maximizing the collecting efficiency by matching the acceptance solid angle of collecting lenses with the numerical aperture of optical fiber. A shutter is installed to protect the collecting window from being coated during the glow discharge cleaning which can cause decreasing window transmittance. The polychromator used for VEST is the same type of KSTAR and is equipped with new set of bandpass filters for measuring $T_e$ in VEST. The number of polychromators will be prepared up to 5 for simultaneous multi-point measurement. As the first stage of development of the TS system, the TS and Rayleigh scattering signals at the single point($R=0.4$m) are measured. We will discuss the result of our measurement and the novel multi-pass optical arrangement scheme.
Development of diagnostic neutral beam injector for charge exchange spectroscopy in VEST

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The combined system of Charge Exchange Spectroscopy (CES) and Beam Emission Spectroscopy (BES) will be developed in Versatile Experimental Spherical Torus (VEST) to measure ion temperature and rotation velocity by not using impurity but fuel hydrogen ion emission line directly. In order to use this system, Diagnostic Neutral Beam Injection (DNBI) system is necessary to supply high energy neutral particles for charge exchange reaction. A 10kW compact neutral beam injection with high-current ion source using electron gun has been developed for VEST. The target beam current of DNB is \(^\sim\)1A at 10kV to get a sufficient light intensity for CES. The beam is injected radially toward center stack to obtain all radial diagnostic data for 20ms operation with 4ms time resolution. The DNBI system consists of four parts: ion source, neutralizer, ion dump with bending magnet and power systems. The electron gun is used in ion source. The ion source can produce high density plasma which is as high as \(^\sim\) for 20ms by being supplied electrons from electron gun. This value shows that the extractable maximum beam current density is . We used triple electrode system with 2mm gap distance to maximize beam current density at low energy. In order to extract \(^\sim\)1A beam current with 4ms modulation, circular hole with 16mm diameter is chosen as extraction hole. A gas flow neutralizer is used to simply structure. For 90% neutralization efficiency, additional gas is injected to 3mTorr at 50cm neutralization region. Considering turbo pump loads, we installed gas tube surrounding beam path and made gas flows only along the tube. In this paper, detailed design of DNB system is presented. And results of diagnostic by calorimeter are also presented.
Development of the bronze processed Nb3Sn multifilamentary wires using Cu-Sn-Zn ternary alloy matrix

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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 beyond ITER. Recently, we approached to the solid solution ternary Cu-Sn (Cu-Sn-X) matrices for the development of the high mechanical strength bronze processed Nb3Sn wires. Generally, it is well known that Zn has the larger solubility limit against the Cu compared with Sn. In the various Nb/Cu-Sn-Zn composite precursors, we confirmed that Zn remained homogeneously into the matrices after the Nb3Sn layer synthesis. The Cu-Sn-Zn ternary alloy was one of the interesting bronze materials for the high strengthened Nb3Sn wires. We fabricated the bronze processed Nb3Sn multifilamentary wires using various Cu-Sn-Zn matrices. In these multifilamentary wires, Zn remained into the matrices after the Nb3Sn synthesis, and then the Vickers hardness of the Cu-Sn-Zn matrices after the Nb3Sn synthesis heat treatment was higher than that of the conventional bronze matrix. Higher Vickers hardness caused by the Cu-Zn phase formation in the matrix. In this study, microstructure and superconducting properties of the bronze processed Nb3Sn multifilamentary wires using various Cu-Sn-Zn matrices were mainly reported. In addition, comparisons of the mechanical properties on the Nb3Sn wires with different Cu-Sn-Zn matrices were also reported. Especially, transport Ic behavior by applying the tensile strain on the Nb3Sn multifilamentary wire using various Cu-Sn-Zn matrices was investigated. This work mainly performed to the Fusion Engineering Research Project (UFFF036) and the collaboration program (KECF013) in NIFS, and also supported by the High Field Laboratory for Superconducting Materials, Institute for Materials Research, Tohoku University (No.15H0024).
A DEMO relevant long leg divertor with external poloidal field coils

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It is accepted that plasma exhaust is a major challenge for DEMO and future power plants and the reference approach is to use a design similar to JET and ITER. There is not yet full confidence this will extrapolate successfully and be compatible with a maximum power flux of 5-10 MWm\(^{-2}\) on the Plasma Facing Components. Detachment provides an attractive solution to the power exhaust problem, radiating power across a large area within the divertor and reducing ion energies below the sputtering threshold of the tungsten targets. Extension of the outer target to a large radius reduces power flux flowing along the divertor leg, diluting the detachment threshold to values compatible with the core. The reduction in power flux with increasing radius also provides a stabilising mechanism for the location of the detachment fount. Scaling the long leg concept up to DEMO relevant machines is often considered impractical due to either excessive loading on coil sets external to the TF or due to the requirement for in-vessel coils. Feasibility of a long leg divertor concept is demonstrated here for a 20.3MA DEMO relevant machine using a set of five PF coils placed external to the TF cage. The outer strike point is extended to 1.5 times the X-point radius without significant modification to the shape of the separatrix. Force, current density and placement constraints are respected across a flat top flux swing of 363Vs. The long leg concept requires a TF coil with a circumference 22\% greater than the reference configuration. The gain in size of the coils and associated structures will undoubtedly increase their cost. However, foreseen ancillary benefits should likewise be considered. These include a reduction in ripple, perhaps enabling a 16 coil configuration, and a reduction in the complexity of remote maintenance schemes.
Experimental stand for thermal-hydraulic tests of forced flow conductors using water at room temperature

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Current models used for thermal–hydraulic analyses of forced-flow superconducting cables used in fusion technology, such as e.g. Cable-in-Conduit Conductors, are typically 1-D and they require reliable predictive correlations for the transverse mass-, momentum- and energy transport processes occurring between the different cable components in order to reliably assess any fusion magnet design in both normal and off-normal operating conditions. The void fraction of some of Nb3Sn superconducting cables designed for the DEMO Toroidal Field coil is strongly reduced (down to about 20%). Conductors with such low void fractions geometries have never been tested for pressure drop yet. Moreover, it was observed that discrepancy between predictions of different bundle friction factor correlations available in literature strongly increases with decreasing the void fraction. There is a need of experimental verification of the accuracy of the existing predictive friction factor correlations at very low void fractions. In this aim a new experimental stand for thermal-hydraulic tests of conductors using water at room temperature is under preparation at West Pomeranian University of Technology, Szczecin. The installation has been designed to enable pressure drop measurements in short samples of conductors with low void fractions in a wide range of Reynolds number. The aim of this paper is to present the new installation and to discuss the first tests results conducted on reference samples and DEMO-like samples fabricated in the EU DEMO design framework.
Analysis of partial blockage of the coolant in a TF coil of the EU-DEMO

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In the European path towards the tokamak reactor DEMO, led by the EUROfusion consortium with the aim of demonstrating electricity production by fusion energy by 2050, the Toroidal Field Coils are under conceptual design. Three different winding pack (WP) options have been proposed by different European parties. In this paper, we consider the ENEA proposal, featuring a layer-wound WP with graded superconductor cross section, in the different layers, depending on the magnetic field. The conductors are obtained from a circular cable-in-conduit conductor (CICC) compacted to a rectangular shape. Two out of six petals are wound around a small spiral, resulting in two low-impedance cooling channels. The WP is encased in a thick stainless steel casing, cooled separately by dedicated cooling channels (CCC) connected to a dedicated refrigeration loop. The 4C code, already successfully used to model the entire DEMO TF coil, is applied here to investigate two off-normal operating conditions resulting from a partial flow blockage in a CICC or from the plugging of one CCC. First, we investigate the effect on the coil performance of a partial collapse of one of the low-impedance channels falling in the most critical WP layer from the temperature margin point of view. The redistribution of the mass flow rate within the WP, driven by the increase in the hydraulic impedance in the partially choked channel, is evaluated. The capability of the other channel to tackle the locally partial choked flow is assessed, when this flow blockage occurs during a plasma burn. Then, the plugging of one of the CCCs in the most critical (plasma facing) side of the casing is analysed, again investigating the impact on the TF coil performance during a plasma burn.
Since the year 2013, the Swiss Plasma Center (SPC) has proposed a Toroidal Field (TF) layout for the DEMO- EUROFusion tokamak, based on a graded winding pack made of layers of Nb3Sn (react-and-wind) and NbTi conductors. In summer 2015, a new reference baseline is issued for the DEMO- EUROFusion tokamak, leading to an update of the TF coil requirements, e.g. the operating current has been reduced from 80 kA to 63 kA. Consequently, the conductor layouts for every graded layer of the TF coil winding pack is re-designed in order to match the new requirements. The each layer of TF coil winding pack has to be connected electrically in series to form the coil. The inter-layer Nb3Sn splice joint design which does not exceed the conductor dimensions is proposed in this paper for the updated 63 kA Nb3Sn TF conductor. This proposed joint design allows a continuous winding of TF coil winding pack from layer to layer, housing the joint at the zone of inter-layer transition. Ultimately, the all inter-layer joints should be arranged within the winding pack at the low-field and low mechanically stressed region of D-shaped TF coil.
A reliable and realistic cost estimate is of paramount importance for the management of large projects, to assist the budget and planning phases. In the case of DEMO, the cost estimate helps driving the selection among competing design options. The achievement of a target construction price $< 2 \text{ B\euro}$ for a 500 MWe fusion power plant is a necessary condition in order to sell electricity to the market price. The magnet system makes up of $\approx 30\%$ of the cost of a fusion device. A cost estimate for the Toroidal Field (TF) coils starts from the unit cost of the basic materials (superconducting strand, copper, steel and insulation), whose market price is quite stable. The cost estimate for the assembly work (cabling, jacketing, winding) is linked to variables like the investment/mortgaging of specialized equipments, the liability of the contractor, the rejection rate and the extent of the Quality Assurance. With the formulae proposed in this work, an estimate of the cost for different designs of the DEMO TF coils is tentatively done, highlighting where a potential exists for cost mitigation in the design and specification approaches. The cost of the Nb$_3$Sn strand remain by far the largest cost driver for the TF coil system.
Three alternative designs of the toroidal field (TF) coil were proposed for the European DEMO being developed under the Eurofusion Consortium. The most ambitious TF coil winding pack in terms of technological deviation from the ITER TF coil design and consequent potential cost saving, the so-called WP1, is based on the react-wind technology of Nb3Sn layer-wound flat multistage conductors. We present the thermal-hydraulic and quench propagation analyses for the WP1 proposed in 2015, in which the realistic magnetic field and nuclear heat load maps, and heat transfer between neighboring turns and layers are taken into account. The aim of the analysis, performed using the Cryosoft software, is to assess the temperature margin at the end-of-burn conditions, as well as the hot-spot temperature and the maximum pressure that is expected in case of quench, and consequently to optimize the WP1 design from the thermal-hydraulic point of view.
Towards a multi-physic platform for fusion magnet design – Application to DEMO TF coil

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In the framework of the EUROfusion DEMO project, studies are conducted in several European institutions for designing the tokamak magnet systems. In order to generate the high magnetic fields required for the plasma confinement and control, the reactor should be equipped with superconducting magnets, the reference design being based on Cable-In-Conduit Conductors cooled at cryogenic temperatures by forced circulation of supercritical helium. In order to propose a TF Winding Pack (WP) design compatible with DEMO requirements, CEA has developed several tools addressing the different areas related to magnet dimensioning. An accurate calculation of magnetic field along the conductor is provided by the TRAPS code, and conductor design is performed using an integrated macroscopic home design code based on simplified models accounting for superconducting properties, mechanics and thermal. This multi-physic tool gives a realistic but not assessed design. Indeed it is based on an assumed operating temperature that must be validated with an elaborated code, since it is linked with temperature margin design criterion (1.5 K). A dedicated modelling tool was developed by coupling the THEA code for 1D thermo-hydraulics in cables and the Cast3M code for 2D transverse thermal diffusion in a limited number of coil cross sections enhancing the accuracy of the outputs as being a quasi-3D approach. This tool allows a better assessment of the flux exchange between WP and casing, and the modeling of inter-turn / inter-pancake thermal coupling. The coupling methodology is described, as well as its validation on an academic case (simulation of a heat exchanger) and its application to CEA proposal for DEMO TF coil. A calculation was performed on the TF coil (CEA configuration) in a burn (steady state) scenario, also considering cooling channels in the casing, and finally showing a safe temperature margin i.e. compliant with 1.5 K criterion.
Central solenoid winding pack design

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The present study aims to minimise the outer radius of the CS coil of European DEMO in order to reduce the size and the cost of the whole tokamak. In a previous study, it has been demonstrated that the outer radius of the CS coil can be reduced maintaining the generated magnetic flux at 320 Vs using high-temperature superconductors (HTS). This first study was based on a uniform current density in the winding pack. In a next step, a layer-wound CS coil with superconductor grading has been considered. It is envisaged to subdivide the winding pack of the CS coil into 10 sections each consisting of a double layer. Depending on the magnetic field at the section in question it is foreseen to HTS in the highest field sections, Nb3Sn at medium fields and NbTi at the lowest fields. Furthermore, a superconductor grading will be implemented in order to reduce the cross-section of superconductor in the lower field sections. A procedure to determine the dimensions of the sections and the relevant cross-sections has been developed including the calculation of the hoop stress in a non-uniform current density winding. An outline design of the winding pack of the CS1 module, experiencing the highest magnetic field, is presented.
Successful operation of Demonstration Reactors is a key step in the fusion development. The structural integrity of the superconducting magnets producing high magnetic fields that are crucial for optimization of a fusion reactor performance must be ensured. Combinations of calculation approaches, reasonable modelling simplifications and clever prioritization at each analysis phase facilitate design optimization by relatively simple and “inexpensive” calculation tools. The mechanical pre-dimensioning of the magnets is extremely useful at an early design stage, prior to the numerical analysis. A calculation tool that reasonably estimates the mechanical strength of the structural components of the toroidal field coil (TFC) system under the dominating electromagnetic forces is described. The novelty of the approach is that it deals not only with the coil casing strength in the critical location but also treats the winding pack wound with the cabled conductor in detail under an essentially 3D stress state. The semi-analytical procedure features optimization of the layered windings by grading the radial and toroidal walls of the conductor jacket separately. The minimum space required for the coil casing and for the winding mechanical structures is defined basing on the strength properties of the pre-selected structural steels. The procedure has been already successfully benchmarked against the numerical solutions for several tentative TFC designs for the European DEMO project. Since the coils pre-dimensioning is limited to the most stressed coil inboard portion, the next analysis step is the 3D FE modelling featuring the homogenized winding. Reasonably simplified and good parametrized numerical models facilitate the sensitivity study and deliver numerical results transparent for the physical interpretation. The detailed “express” conductor analysis based on the calculated averaged winding stresses is attributed to this analysis stage. Critical winding locations are found as the worst combinations of conductor stress components. Examples from a current fusion project are given.
The tilted toroidal field coil concept for advanced tokamaks

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Tokamak toroidal field coils (TFCs) characterized by a tilting in the azimuthal direction lead to several potential advantages, most notably the relieving of the stresses in the most critical area at the inboard side. As a consequence, much of the heavy steel structures needed to withstand the huge electromagnetic forces in conventional magnets can be reduced. Mechanically unloading the TFCs makes it easier to generate the fields required to approach ignition conditions at high density and relatively low temperature in compact devices. An additional advantage of the tilted coil is that of generating poloidal field as well as toroidal field. If the former is used to provide at least some of the flux swing needed to induce the plasma current, then the discharge could be sustained for a longer time. The “tilted coil” concept originates from an idea presented almost thirty years ago in two papers [1,2] dealing with the problem of generating large toroidal fields in compact tokamaks. In perspective, tilted coils could be made of ribbons of high temperature superconductors, characterized by high critical magnetic fields but rather poor structural properties. The magnetic field produced by a system of tilted TFCs of different shapes (rectangular, circular, D-shaped) has been simulated with an ad hoc numerical code, and an optimization procedure was implemented to find the tilting angles that minimize the total force or its components in some direction. It is found that the radial force on the inner leg can be reduced by over a factor of ten, while in the remaining regions of the coil the forces are reduced by a lesser extent, and their direction change. [1] A. Sestero, Comment. Plasma Phys. Controll. Fusion 11, 27 (1987). [2] B. Coppi, L. Lanzavecchia, et al., Plasma Phys. Controll. Fusion 11, 47 (1987).
P1.092

Conceptual design study of toroidal field magnet system of SST-2 fusion reactor


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SST-2 is a medium size fusion reactor machine under design at Institute for Plasma Research, India. It is being planned to operate between 100-300 MW of fusion power with main objectives of breeding of Tritium, Tritium handling studies and as a test bed for materials and components. SST-2 physics requirements of toroidal field $B_t = 5.42$ T at plasma major radius $R = 4.42$ m and the maximum allowable magnetic field ripple of less than 1% at the last close flux surface (LCFS) have been used as the design driver of Toroidal Field (TF) magnet system. In addition to these requirements, the available accessibility at the outboard side for maintenance activities and the neutral beam port size requirements dictate the overall size of the TF coil. This paper will give details of studies done for TF coil shaping, sizing and the electromagnetic analysis for the SST-2 TF coil magnet system.
The preassembly of the tokamak T-15MD magnet system

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Presently, the Tokamak T-15MD (T-15U) is being built. All elements of the magnet system have been manufactured by the end of 2015. The magnet system of the Tokamak T-15MD will obtain and confine the hot plasma in the divertor configuration. The tokamak T-15MD magnet system includes the toroidal winding, the poloidal magnet system and supporting structures. The toroidal winding consists of 16 D-shaped coils forming the arched structure. The poloidal magnet system includes a central solenoid, poloidal field coils and horizontal field coils. PF coils are placed outside the toroidal winding and are fastened to the TF coil cases. The four framed form horizontal field coils are placed around the torus in the space between the vacuum chamber shell and the toroidal winding. The preassembly of the tokamak T-15MD magnet system is conducted at plant in Bryansk. The purpose is the conjunction of all elements among themselves, obtaining the necessary experience for later tokamak assembling in Kurchatov Institute. The results of preassembly of the tokamak T-15MD magnet system are presented.
Joint testing of the 3 Tesla ST40 spherical tokamak toroidal field coil test assembly

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Spherical Tokamaks used in magnetic fusion have a small centre stack by design. This causes a very high field on the conductor. ST40 is a 3 Tesla spherical tokamak with a major radius of R=40cm and minor radius of a=26cm being built by Tokamak Energy. The high toroidal field (TF) requirement requires a wire current of 250kA flowing in each of the 24 limbs totalling 6 MA in the centre stack. Joint testing was used to investigate the interface resistance between the centre stack and return limbs under a variety of contact pressures, using different shims, different coatings on the conductor and soldering. Initially a DC 400A current source was used and later an air-core pulsed transformer delivering up to 16kA of current was specifically designed and built for these measurements. Results from this paper can be used to predict the behaviour of critical joints in the region at the ends of the centre stack where current is passed to the TF return limbs. A comparison with TF joint data from the National Spherical Torus Experiment (NSTX) is also given. This will aid in deciding which jointing method to use for ST40 and the required pressure to pre-load the interface.
High-Temperature Superconductor (HTS) material REBCO has high critical currents even in high magnetic fields. The use of such material for future fusion magnets was already proposed in 2004, but the aspect ratio of REBCO, which is available as thin tapes only, made the realization of a high current cable in the current range of several 10 kA at magnetic fields around 12 T difficult. In the last years a number of cabling proposals were made, including HTS cable demonstrators reaching fusion relevant currents in the appropriate magnetic field, e.g. a cable demonstrated by SPC reached 60 kA@12 T using round twisted stacked strands. A relative new proposal refining the SPC approach is the formation of the HTS CrossConductor (HTS CroCo), using a cross shaped arrangement of REBCO tapes with two different widths to optimize the engineering critical density in an outer circular shape. Adding an outer envelope, this approach aims for an easy long length production of a round HTS CroCo unit, which may either be used for energy efficient power transmission in self field condition or as a basic strand for a high current cable in large high field coils. The poster will outline the HTS CroCo approach and show new results on an HTS CroCo fully equipped with 6 and 4mm REBCO tapes measured in the FBI facility.
In the last years, W and W-Ti and W-V alloys, with grain sizes of hundreds of nanometers and densification very close to 100%, have been produced following a powder metallurgy route that consists of mechanical alloying and consolidation by hot isostatic pressing (HIP). In spite of the submicron-grained microstructure, and the dispersion of second phase nanoparticles, these alloys do not exhibit a significant ductility enhancement with respect to pure tungsten. A relevant microstructural feature in the W-4Ti alloys that might give account for the mechanical behavior is the Ti segregation in large pools and dispersed particles with a wide range of sizes. The same has been found for the W-V alloys. The goal of the present study has been to study the size distribution of the second phase nanoparticles dispersed in W-Ti and W-V alloys using the Small Angle Neutron Scattering (SANS) technique. Unlike other techniques, as Electron Transmission Microscopy (TEM) or atom probe tomography (APT), SANS allows to obtain information about a macroscopic volume of the material. W-xwt%Ti and W-xwt%V alloys (x=2 and 4) have been produced by mechanical alloying and HIP. The SANS experiments were carried out in the KWS-2 spectrometer of the FRMII at Garching (Germany) using a wave-length of 7 Å. The analyses of the scattering curves in terms of a polydisperse particle system using the Beaucage approach suggest that the particle dispersions may consist of two structural levels, or particle populations. The experimental data have also been analyzed by the maximum entropy method.
P1.103

Potential irradiation of Cu alloys and tungsten samples in DONES

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Tungsten and Cu-alloys are currently proposed as reference candidate material for ITER first wall and divertor. Tungsten is proposed for its high fusion temperature and Cu-Cr-Zr alloys for their high thermal conductivity together good mechanical properties. However its behavior under the extreme irradiation conditions as expected in ITER or DEMO is still unknown. Due to the determinant role of H and He played in the material behavior any irradiation experiment must take into account the important amount of these gases produced during the irradiation in Fusion reactors. DONES (DEMO Oriented Neutron Source) has been conceived as a simplified IFMIF-like plant to provide in a reduced time scale and with a reduced budget – both compared to IFMIF- the basic information on materials damage. The objective of DONES-IFMIF in its first stage will be to test structural materials under similar neutron irradiation nuclear fusion conditions as expected in fusion reactors. These tests will be carried out in specimens irradiated in the so-called High Flux Test Module (HFTM). The objective of this paper is to assess on the potential use of DONES to irradiate Cu alloys and tungsten in the HFTM together other stainless steel based materials. The presence of Cu alloys or w specimens may have an effect in the irradiation parameters of the stainless steel samples placed also in the HFTM and in the samples of the Creep Fatigue Test Module (CFTM). Alternatively the Cu alloy specimens could be located in the volume made available in DONES with the absence of other irradiation modules. These different locations are analyzed and compared. McDeLicious code is used for neutron transport calculations. Damage dose rate and H and He production are analyzed in the different locations and compared with the actual irradiation conditions in first wall and divertors in fusion machines.
The exhaust of power and particles is regarded as a major challenge in view of the design of a nuclear fusion demonstration power plant (DEMO). In such a reactor, highly loaded plasma facing components (PFCs), like the divertor targets, have to withstand both severe high heat flux (HHF) loads and considerable neutron irradiation. Existing divertor target designs, as e.g. the ITER-like monoblock concept, make use of monolithic W and Cu material grades that are combined to a PFC. Such an approach, however, bears difficulties as W and Cu are materials with inherently different thermomechanical properties and their optimum operating temperature windows do not overlap. Against this background, W-Cu composite materials are promising candidates regarding the application to the heat sink of highly loaded PFCs. In principle, these materials feature a high thermal conductivity combined with acceptably ductile behaviour due to a coherent Cu or Cu alloy matrix. Moreover, they exhibit strength properties significantly higher compared to the used matrix material due to the presence of the W inclusions. Above all, W-Cu composite materials offer metallurgical flexibility as their macroscopic properties can be tailored by customising their microstructure. The contribution will present the latest results regarding the industrially viable manufacturing and characterisation of W-Cu composite materials produced by means of liquid Cu melt infiltration of open porous W preforms. On the one hand, this includes composites manufactured by infiltrating powder metallurgically produced W skeletons. On the other hand, W-Cu composites based on textile technologically produced fibrous reinforcement preforms are discussed. Furthermore, it will be pointed out how these materials can be integrated into PFCs.
Joining of armor material tungsten to other alloys and especially to copper components which will act as heat sinks in divertor application showed lacks due to the restricted miscibility of tungsten and copper. This negative behavior leads to bad or missing metallurgical W – Cu reactions with the consequence of reduced mechanical stability or high risks of cracking if any joining was realized. Introducing adapted interlayers can overcome these limitations if they exhibit some extended miscibility with both parts to be joined as, e.g., the elements Fe, Cr, Ni or Pd. Electrochemical plating was chosen as deposition technology for such reactive interlayers and the plating characteristics of preferably Pd on W and Cu was analyzed to obtain adherent and industrially applicable coatings. The electrochemical plating was performed applying an aqueous Pd electrolyte based on an ammonia complex. Based on this evaluated plating parameters demonstrators were processed with a 10 µm thick reactive Pd interlayer and joined by diffusion bonding. Their metallurgical behavior was characterized in dependence on processing temperature, reaction time and applied pressing load. Beyond of analyzing the reactive joining process the fabricated joints were mechanically qualified by shear testing. Cracking of the joints never appeared at the boundary of interlayer to W which was the critical zone of pure W – Cu connections. The demonstrators revealed reasonable and applicable shear strength of around 100 MPa. The observed shear strength values and formed microstructures in the joining zone will be displayed and discussed in dependence on the applied processing parameters. The developed bonding process by applying electrochemically plated interlayers has proven to be a reliable tool with industrial application potential.
Design of high-strength, high-conductivity, creep-resistant Cu alloys for fusion high heat flux structures

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Although high room temperature strength (300-1000 MPa) and conductivity (200-360 W/m-K) have been achieved in Cu alloys, these alloys suffer significant thermal creep deformation at temperatures above 300-400°C. Deformation analysis indicates dislocation creep and grain boundary sliding are occurring. Design requirements for improved high-performance copper alloys are: 1) thermally stable microstructure up to high temperatures; 2) precipitates to inhibit creep deformation (grain boundary sliding) that are stable under neutron irradiation; and 3) sufficient sink strength to enable suitable radiation resistance. Creep resistance can be improved by using relatively large particles along grain boundaries to inhibit grain boundary movement, along with a high density of fine-scale matrix precipitates to suppress dislocation motion. The fine-scale matrix particles also provide beneficial radiation resistance. For high creep strength, high thermal conductivity and radiation resistance, the optimized matrix precipitates should have a volume concentration near 1-5% with an average particle diameter near 10 nm. The matrix and grain boundary particles must be resistant to thermal and radiation-enhanced coarsening during extended times (>1 year) at operating temperatures. These particles should also be thermally stable during short-term exposure to joining-relevant temperatures (brazeing, HIP, etc.). Computational thermodynamic calculations of Cu-Cr-Zr-based quaternary alloys have identified several promising compositions using conventional metallurgy processing to produce a bimodal distribution of large grain boundary particles (Cu5Zr and laves phase) and a high density of matrix precipitate particles (e.g. Cr precipitates) that can be aged to provide good matrix strengthening up to 400-500°C. Small research heats of these newly designed high conductivity creep-resistant Cu alloys have been fabricated. Microstructural characterization is being performed on the as-fabricated alloys and after several heat treatments to examine the overall distribution and thermal stability of the precipitates. The results of elevated temperature tensile and thermal creep tests to quantify the mechanical properties will be summarized.
Residual stresses evaluation during Plasma Facing Units
Hot Radial Pressing manufacturing process

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The ITER operation program, as well as the DEMO operational, foresees for the vertical targets strike point region high steady state thermal fluxes that can be sustained only by components designed and manufactured accordingly. Their life-time is limited mainly by thermal fatigue caused by cyclic thermal loads inducing high mechanical stresses. The Plasma Facing components of the ITER divertor are made of a CuCrZr tube joined to tungsten drilled rectangular blocks (named monoblocks) provided of a complaint layer of pure copper (Cu-OFHC). In the framework of an EFDA contract (05-1249), ENEA manufactured six small mock-ups of the ITER divertor Inner Vertical Target plasma facing units that were tested to thermal fatigue loads at High Heat Flux (HHF) in the TSEFEY e-beam facility at the Efremov Institute. A comparison between FEM simulation and HHF test results has been presented in a previous work, where the lifetime estimated was less than that found during the experiments. A better estimation of the test results can be obtained by including in the calculation the residual stresses induced in the CuCrZr tube by the manufacturing process of the mock-ups. The reference manufacturing process used for the calculation is the Hot Radial Pressing (HRP). This process developed by ENEA and widely qualified is based on the radial diffusion bonding principle performed between the cooling tube and the armour blocks. The bonding is achieved by pressurizing internally the cooling tube while the joining interface is kept at the vacuum and temperature conditions. In the simulation, the joining process is considered and the residual stresses are the initial condition for the subsequent simulation of the HHF testing. Calculation of the mock-ups lifetime under high thermal loads fatigue is performed and the results are presented and again compared with the HHF test results.
Influence of divertor material modifications on the inventory of tritium in the divertor region

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Fusion device materials have been modified over the years for the main aim of using optimal materials in ITER fusion device. Post-mortem analysis of materials used in JET provides valuable information for further material development and improvements required. One of key fusion device elements is the divertor. It minimizes plasma contamination and draws a big part of thermal and neutron load in the fusion device. Over the years it was proven that CFC material itself is not an outstanding divertor material. It has high tritium affinity and plasma erosion rate. Therefore W coatings have been applied to test the efficiency of W in reducing tritium accumulation in the fusion device divertor region. In the research various divertor tiles from JET 2007-2009 and 2010-2012 campaigns with W coating have been analysed for tritium accumulation depth profiling. A comparison to profiles of tritium accumulation in uncoated tiles of comparable divertor positions was also done. The results show gradual decrease in tritium activity on the surface of divertor tiles when W coatings are applied (approximately \(10^5\) Bq g\(^{-1}\) for W-coated samples compared to \(10^6\) to \(10^8\) Bq g\(^{-1}\) for various uncoated samples). Though efficiency of W coatings in reducing the amount of accumulated tritium varies considerably and does not depend strictly on W coating itself. Samples of tiles 7 and 8 that are not in constant direct interaction with plasma show smaller difference between surface activity (\(10^4\) Bq g\(^{-1}\)) and bulk activity (\(5\times10^4\) Bq g\(^{-1}\)) than samples of tiles closer to Louvre (\(10^6\) Bq g\(^{-1}\) on the surface, \(10^4\) Bq g\(^{-1}\) in the bulk). As it can be concluded, the W coating is efficient in reducing inventory of tritium in divertor region, though it varies strongly on the tile position and plasma interaction.
Investigation of hydrogen isotopes interaction with lithium CPS under reactor irradiation

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Application of liquid lithium as a plasma facing material has some features proved by a lot of experiments with lithium devices in plasma accelerators KSPU, MK-200UG and “Plasma focus” facility. Then, the experiments carried out in operating tokamaks and stellarator (NSTX, FTU, T11-M, EAST, TJ-II) using liquid lithium and lithium CPS as intrachamber devices have shown the advisability of lithium application and its attractiveness compared to traditional materials. In continuation of this, the idea of new composite material based on capillary-porous system made from different metals and alloys filled by liquid lithium, was proposed in Russia. This material has a self-retainable surface with self-regulating lithium consumption. One of the problems connected with application of such liquid lithium systems in fusion reactors is determination of interaction parameters of plasma facing surface with working gases under conditions of fusion devices real operation, e.g. under neutron and gamma radiation. This work presents the experimental results on study of hydrogen isotopes interaction (including tritium) with lithium CPS samples under neutron irradiation and without it. The experiments were performed at the IVG1.M research reactor (Kurchatov, Kazakhstan) at different reactor power levels and temperatures from 473 to 773 K. In studies the data on temperature dependencies of tritium release rates from lithium CPS samples under different deuterium pressures over sample were obtained.
Erosion and morphology changes of F82H steel under simultaneous hydrogen and helium irradiation

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The use of bare Reduced Activation Ferritic Martensitic (RAFM) steels has been proposed for the first wall in a reactor \([1]\). Thus, it is necessary to understand the performance of RAFM steels under fusion-relevant condition. To date, the effects of simultaneous irradiation of hydrogen isotopes and He in F82H haven’t been examined in detail. We previously examined hydrogen retention properties, and reported systematic reduction due to simultaneous D+He irradiation \([2]\). In this contribution, we report on the sputtering behavior and accompanying surface morphology changes. Experiments were performed over the temperature range of 500-1000 K, consistent with the anticipated operating temperatures of the blanket region. Simultaneous H+He or H-only irradiation was performed with 1 kV extraction voltage. The irradiation fluence was \(1 \times 10^{24} \text{H/m}^2\). He % in the ion beam was 0.5 %. Mass loss was measured before and after irradiation to determine erosion. Surface morphology changes were observed by SEM/FIB. RBS was used to quantify the near surface W concentration at NIFS. From SEM images, surface roughening was observed at T > 750 K in both H-only and H+He irradiations. The effect of He was minor. At 865 K, the roughness increases dramatically with the peak to peak differences in the order of \(\sim \mu \text{m}\), which is much greater than the ion implantation zone ( <10 nm). Correspondingly, the sputtering yield of F82H increases with increased surface roughening. However, the sputtering yield for H+He irradiation was systematically higher compared to H-only irradiation. This is attributed to lower W surface enrichment at the surface as confirmed from RBS measurements. This work is performed with the support and under the auspices of the NIFS Collaboration Research Program (NIFS15KEMF072) \([1]\) D. Maisonnier et al., Fusion Engineering and Design 75-79 (2005) 1173-1179 \([2]\) K. Yakushiji et al., Physica Scripta T167 (2016) 014067
Implementation of ferritic steel as in vessel wall: Lessons learnt and followed up

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ASDEX Upgrade (AUG) is the only tokamak in Europe to have low activation ferritic steel in the inner vessel wall. The project is a first step towards the extensive use of ferritic steel in future fusion reactors. The ‘ad hoc’ ferritic steel built with low activation capability is the so called Eurofer. As the low activation property is not a requirement for AUG, the material selected for the project is the martensitic steel P92 which is the most similar material to Eurofer from a magnetic point of view. The purpose of the project is to improve understanding of the magnetic perturbation of the ferritic steel both on the plasma and magnetic probes, evaluating and controlling these effects. Bearing this in mind, in 2013 a step wise program has been started and part of the graphite tiles in the region of the inner column with steel tiles were replaced [1]. The first campaign did not suffer any particular problem related to the new material, but the inspection of the machine pointed out some hardware problems. The graphite tiles adjacent to the steel tiles were damaged. To justify the failure mode inside the vessel, a hypothesis was made and FEM analyses were carried out in this direction. With extreme caution, in 2015 just one additional steel row was added together with diagnostics that confirmed the hypothesis. Now that a clear understanding of the problem has been reached, the project could be continued. For the next campaign it is planned to replace all the tiles in the middle region of the heat shield together with stiffening and modification of the supporting structure. In this paper the learning process from the damage of the tiles and its causes, from the FEM analysis results to the data diagnostics will be reported.
Effect of engineering constraints on charged particle wall heat loads in DEMO

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The design of the demonstration fusion reactor DEMO presents challenges beyond those faced by the ITER project and may require the implementation of different solutions. One of the biggest challenges is managing the heat flux to the main chamber wall. The presently predicted total heating power in DEMO is more than 3 times that predicted for ITER value, while the major radius is only 1.5 times greater. Furthermore the present DEMO technological wall heat load limitation is limited to ~1MW/m², due to structural material limitations and the tritium breeding requirements, while the ITER first wall is designed for values up to 4.7MW/m². This paper focuses on the evaluation of the effect of the engineering constraints on the charged particle heat load. First, a series of optimizations on the plasma and first wall 2D shape is presented. A subset of the resulting configurations was analysed using the 3D field line tracing code PFCflux, to derive the heat flux on to a 3D engineering model of the first wall. Finally a sensitivity scan was performed on the main wall design geometrical parameters and on manufacturing and installation tolerances. The resulting heat flux peaking factors were up to a factor ~10 leading to a value of the heat flux on the wall up to 10MW/m². While some optimization can be reached with detailed shaping of the components, this is in line with extrapolation from the ITER values. These technological limitations of DEMO, i.e. the ~1MW/m² limit, may push towards the adoption of discrete high heat flux limiters, with the implications on their remote maintainability and the breeding capability to be analyzed. The developed methodology will be used to efficiently include and prescribe the manufacturing and installation tolerances for the DEMO components as they become available.
Effect of ELMs on PFC of DEMO reactor

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The first assessments has shown that the edge localized modes (ELM) in the fusion power plant DEMO will pose a severe tread to the plasma facing components (PFC) by causing a surface melting and erosion [1]. In this work we estimate the degree of the ELM mitigation required for sustaining of W-armour during operation in the temperature range above ~700°C and below the recrystallization temperature ~1300°C. The characteristics of the edge localized modes in the next step reactor are re-estimated and derived by using the scaling arguments found in experiments, the prediction of peeling-ballooning mode theory and by extrapolating results made for ITER. The tungsten armour damage of the PFC pressured water cooled module due to the repetitive ELM loads is numerically investigated by using the MEMOS code [2]. It is shown that for the controlled ELMs a pressured water reactor conditions in DEMO can ensure an effective cooling and prevent materials of the PFC modules from distortion.

Integrated core-SOL-divertor modelling for DEMO with tin divertor

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The DEMO device is expected to operate in H-mode. On the other hand it is postulated that the divertor power load cannot exceed 5MW/m\textsuperscript{2}. In case of liquid divertor, vaporizing additionally enhances the plate material flux into the bulk. Impurities with large atomic number (Z) dilute the plasma core less, however, they radiate more in the core than those with smaller Z. Liquid tin (Sn) or lithium (Li) divertors are considered as alternatives to a standard tungsten (W) one. This paper analyzes a possible operational space for the DEMO device with the liquid tin (Sn) divertor setup. The Sn (Z=50) impurity originating from the sputtering and vaporizing is expected to modify plasma characteristics significantly both in the bulk and in the scrape-off layer. 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 the sputtering, prompt redeposition and evaporation of the liquid Sn divertor is taken into account. An operational space of parameters for power to SOL higher than the L-H threshold and the power to the plate less than the technological limit is found. First simulation without impurity seeding shown, that plasma in DEMO with the Sn divertor characterizes with 84\% radiation fraction. However, power to the plate is about 55 MW, which is higher than the limit. In order to reduce power to the plate neon and argon seeding were included. Preliminary results show that in case of Ar seeding power to the plate can be reduced to about 20 MW and in case of Ne seeding reduction to the level of 14MW might be achieved.
Physics design study of the divertor power handling in 8m-sized DEMO reactor

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Handling of the huge power exhausting from the core region to the SOL/divertor region is one of the crucial issues for a DEMO reactor design. In previous study for JA compact DEMO concept, SlimCS (a major radius of 5.5m), numerical simulation by an integrated divertor codes SONIC showed the divertor target heat load of < 10 MW/m² for the fusion power of < 1.5 GW and the large impurity radiation fraction on the exhausted power (frad > 0.8) by the Ar gas seeding. Recently, JA-Model 2014 DEMO concept (a major radius of 8.5m, a plasma current of 14MA, a fusion power of ~1.5GW) has been proposed. In the concept, the operational density becomes low compared with previous concept SlimCS, due to the low Greenwald density of ~6.6x10¹⁹ m⁻³. The divertor power handling scenario with the divertor plasma detachment may become difficult in such low operational density. In this paper, a divertor power handling scenario with the detachment for the 8m-sized DEMO is studied by using SONIC code. Even in the case of the low SOL density of ~1.5x10¹⁹ m⁻³ at the outer mid-plane, the partial detachment at the outer divertor target is obtained due to large machine size and large frad of 0.8. The SONIC simulation shows the target heat load of ~7 MW/m². However, the divertor plasma is still attached at the region far from separatrix and the peak ion temperature exceeds 300 eV, which causes the significant target erosion. Effects of increasing the SOL density by the fuel gas puff, location of the gas puffing, the divertor geometry effects, etc. on reduction of the ion temperature are also studied.
Progress in the initial design activities for the European DEMO divertor

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After the preliminary exploring phases for devising initial design concepts and performing design studies, the divertor project (WPDIV) of the EUROfusion consortium is currently entering into the final stage of the first half R&D round which is planned to be completed by the end of 2016. The core missions of WPDIV are to deliver feasible pre-conceptual design solutions for the divertor of an early DEMO reactor, to develop key technological elements needed to materialize the design concept and to verify the required high-heat-flux performance of the selected target concepts. In WPDIV two subprojects were installed: ‘Cassette design and integration’ and ‘Target development’. In the subproject ‘Cassette’ the overall system architectures of a cassette body are engineered whereas in the subproject ‘Target’ advanced design concepts and technologies are developed for the plasma-facing components. In this contribution, a brief overview is given on the recent progress achieved in the R&D activities. Major progress to be highlighted in this presentation includes: the thermo-hydraulic design of cassette cooling schemes, multi-physics analysis of cassette loads, cassette interior design, CAD models for fixation supports and coolant piping, novel design concepts for the plasma-facing target components, modelling of structural failure, engineering solutions for mock-up fabrication, non-destructive inspection of mock-ups and the first high-heat-flux test campaign (if available by then). Finally, impact of the divertor design requirements on the R&D programs of the related work packages (e.g. high-heat-flux materials, structural design rules, remote handling, etc.) is discussed.
P1.117

Advances of the Design Study of ITER-like divertor target for DEMO

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DEMO development is currently in the Pre-Conceptual Design Activity and the Divertor that is in charge of power exhaust and removal of impurity particles represents the key in-vessel component, with its Plasma Facing Units (PFU) exposed to the plasma and hence subjected to very high heat loads. During 2015 the integrated R&D project launched in the EUROfusion Consortium studied how to approach and solve the many issues that DEMO will have to face: in fact unlike the ITER machine, DEMO will be subjected to an intense neutron irradiation that will cause damage and defects in the materials due to operation almost stationary during the envisaged lifetime of two full power years. This paper deals with the advances in the design study of an “optimized” ITER-like Water Cooled Divertor able to withstand a stationary heat flux of $10\text{MWm}^{-2}$, as requested for DEMO operating conditions. The structural material for the heat sink pipe made in CuCrZr was assessed by means of a dedicated computational analysis procedure selecting appropriate ITER SDC-IC rules (3Sm and fatigue), together with the thermal margin to the local critical heat flux (CHF) at the cooling tube and the windows operating temperature for the material itself. The design rationale for the PFUs will be provided too. Further activities within the Eurofusion program foresee mock-up fabrication and high-heat-flux (HHF) tests.
Development of a graded W/CuCrZr divertor for DEMO reactor

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The divertor is the key in-vessel plasma-facing component being in charge of power exhaust and removal of impurity particles. The operational reliability of the divertor target relies essentially on the structural integrity of the component, in particular, at material interfaces, where thermal stresses tend to be concentrated and thus cracks are most likely to initiate. In this context, the quality of material joining is of crucial importance and simultaneously a technological challenge. One fabrication option to reduce the thermal stress at the bond interface of a joint component is to insert a functionally graded interlayer (FGI) into the interface. In this paper, recent development of a novel monoblock type target concept is presented where the tungsten armor monoblock is joined to the copper alloy cooling tube via a thin functionally graded W/Cu interlayer. First results of the development activities are reported focusing on the fabrication process, geometry optimization and heat exhaust performance. The FGI was manufactured using plasma vapor deposition (PVD) and the final assembly was made using hot isostatic pressing. Due to the limitation in the deposition rate of PVD process, the FGI thickness was chosen to be 15 µm. The chemical composition, thickness and the adherence strength of the FGI samples were characterized and found to fulfill the material requirements. A couple of test mock-ups were manufactured on the basis of the optimized geometry which was identified by finite element analysis applying selected structural design criteria. The bonding quality of the fabricated joint mock-ups measured by infrared and ultrasonic inspection is also addressed. Finally, the result of high heat flux tests conducted to evaluate the heat exhaust capability of the mock-ups is presented together with the computational prediction.
Analysis of steady state thermal-hydraulic behaviour of the DEMO divertor cassette body cooling circuit

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In the framework of the work package “Divertor” of the EUROfusion action, a research campaign has been jointly carried out for the subproject “Cassette design and integration” by ENEA and University of Palermo to investigate the thermal-hydraulic performance of the DEMO divertor cassette body cooling system. A comparative evaluation study has been performed considering the different options of divertor cassette body coolant and namely pressurized water and helium. The research activity has been carried out following a theoretical-computational approach based on the finite volume method and adopting a qualified Computational Fluid-Dynamic (CFD) code. A realistic finite volume model of the cassette body cooling circuit has been set-up and a sensitivity analysis has been carried out in order to select a mesh fine enough to give stable predictions without inducing too long calculation times. The k-ε turbulence model, typically suggested for general purpose simulations and offering a good compromise in terms of accuracy and robustness, has been adopted for the calculations and the automatic scalable wall functions have been used to simulate the near-to-wall region flow. CFD analyses have been carried out for the considered options of cassette body cooling circuit under nominal steady state conditions and their thermal-hydraulic performances have been assessed in terms of overall coolant thermal rise, coolant total pressure drop, flow velocity, pumping power and wall heat transfer distribution at the fluid-structure interface, to check whether they comply with the corresponding limits or to give them in input for the numerical assessment of the cassette body thermo-mechanical performances. Results obtained are reported and critically discussed.
Thermal-hydraulic behaviour of the DEMO divertor plasma facing components cooling circuit

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In the framework of the work package “Divertor” of the EUROfusion action, a research campaign has been jointly carried out for the subproject “Cassette design and integration” by ENEA and University of Palermo to investigate the thermal-hydraulic performance of the DEMO divertor cassette cooling system. A comparative evaluation study has been performed considering three different options of cooling circuit layout for the divertor Plasma Facing Components (PFCs). The potential improvement in the thermal-hydraulic performance of the cooling system to be achieved by modifying the coolant circuit layouts has been also assessed and discussed in terms of optimization strategy. The research activity has been carried out following a theoretical-computational approach based on the finite volume method and adopting a qualified Computational Fluid-Dynamic (CFD) code. CFD analyses have been carried out for the PFCs cooling circuit lay-out options under nominal steady state conditions and their thermal-hydraulic performances have been assessed in terms of overall coolant thermal rise, coolant total pressure drop, flow velocity and CHF margin distributions along the vertical target Plasma Facing Unit (PFU) channels, to check whether they comply with the corresponding limits. Results obtained have clearly predicted very modest coolant thermal rises (lower than 10 °C) for all the PFCs cooling options investigated as well as a sufficient margin against CHF onset (higher than 1.4) along all their PFU channels. Conversely, estimated total pressure drops have resulted higher than the limit of 1.4 MPa for all the PFCs cooling options investigated, especially in case of option 2. Therefore, an optimisation study has been carried out to minimize the cooling options total pressure drop by properly changing their geometric configuration. In particular, the potential effect of increasing PFC inlet/outlet manifold diameter has been investigated with encouraging results for all the three options. Results obtained are reported and critically discussed.
Electromagnetic and structure analysis for EAST vacuum vessel with plasma facing components during VDE

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East Advanced Superconducting Tokamak (EAST) is a superconducting magnet tokamak and its goal is to achieve the magnetic confinement fusion. The major plasma disruption (MD) or the vertical displacement event (VDE) all will produce toroidal eddy current in the vacuum vessel (VV) with plasma facing components (PFCs) and cause mechanical forces, which represent one of the most vital loads for tokamak. This paper is focused on calculational methods and results for the electromagnetic loads on the simplified but practical model of EAST VV with PFCs respect to VDE scenarios based on outputs from DINA, which are one of major sources of electromagnetic loads on VV and PFCs. Commercial finite element method software, ANSYS, was employed to evaluate the eddy current on the VV and PFCs modules with the 22.5 degree sector model for major conducting structure of the tokamak. As the results of calculating the eddy currents and electromagnetic forces, stress and deformation on EAST VV with PFCs can be obtained. According to the analysis results, some advices to more effectively protect EAST vacuum vessel and PFCs from being destroyed in EM event is given out.
P1.122

Progress of engineering design and analysis of hl-2m advanced divertor

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A medium sized Tokamak HL-2M is being designed and constructed in Southwestern Institute of Physics of China. This device can be operated with high plasma current 2.5 MA and toroidal magnetic field 3 T. Advanced divertor configurations with snowflake, tripod etc. are envisaged to study the divertor physics under high heating power and high core plasma performance operation. To accommodate the various advanced divertor configurations, a robust and flexible engineering design of divertor is expected. In this paper the design and analysis progress of HL-2M divertor has been introduced. The divertor assembly is divided into 80 modules to facilitate the installation, maintenance, and local upgrade. Each module is mainly composed of CFC tiles, tile carriers, support box beam, and cooling pipes, and these components are integrated by the support beam via hinge/bolt joints allowing thermal expansion and tolerating clearances. CFC tiles as the plasma facing material are brazed on the tile carriers which are drilled inside with channels to feed cooling water during plasma discharge and hot nitrogen during 300°C baking. These channels are connected to water pipes embedded inside the curved box beam to minimize the risk of plasma bombardment. High precision support ring and flexible connection with vacuum vessel are designed to install the divertor assembly precisely. Two-phase thermal hydraulic analysis has been done to research the cooling performance, and structural integrity analyses have also been done based the thermal gradients and electromagnetic loads. The results show that HL-2M advanced divertor can survive under 10MW/m\textsuperscript{2} cyclic operation and satisfy the requirements from physicists.
Cooling concepts for CFETR divertor target

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The China Fusion Engineering Testing Reactor (CFETR) aims at bridging the gap between ITER and DEMO. Its scientific mission is to produce fusion power of 200 MW with tritium self-sustention and duty cycle of 0.3-0.5. The big fusion power and the auxiliary heating power of 100-140 MW, makes the design of CFETR divertor challenging. Previous work focuses on the plasma configuration and the first round engineering conceptual design, in which the divertor target employs the ITER-like water-cooled W/Cu monoblock. However, this W/Cu concept is only feasible for the operation phase I when the neutron dose level is comparable with ITER. While in operation phase II, the neutron dose level is much higher, evaluated as 5 dpa/year in the divertor. As a result, the high activation of CuCrZr heatsink prevents the use of W/Cu concept. Therefore, new cooling concepts have been studied. The first updated one is still based on the W/Cu concept, whereas the CuCrZr is replaced by the China Low Activation Martensitic steel (CLAM). Unfortunately the low thermal conductivity of CLAM, ~28 W/(mK), drastically decreases the heat loads capability. After optimization of geometrical parameters of the monoblock and introducing a felt-metal as interlayer between the tungsten and the CLAM, with proper hydraulic parameters the structure can afford 10 MW/m² heat flux in steady state. In addition, a novel concept was proposed that with tungsten alloy WL10 as heatsink and molten salt (FLiNaK or FLiBe) as coolant. The initial designed divertor target is a 5 mm thick tungsten tile brazed onto a 1 mm thick filleted rectangle WL10 heatsink. Based on thermo-hydraulic and mechanical calculations, with proper hydraulic parameters the design can sustain steady state heat loads higher than 10 MW/m². The detailed design and main calculation results are presented in the paper.
P1.124

Evaluation of heat load on CFETR divertor for impurity seeding using SOLPS

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The Chinese Fusion Engineering Test Reactor (CFETR) is under design. Divertor is the most pivotal PFC to manage power and He ash exhaust. Based on the main goal of CFETR, it has a similar P/R ~ 14 MW/m to ITER. Impurity seeding has been considered a promising means to enhance the radiation from the plasma edge and hence to reduce the target heat load, especially on carbon-free wall conditions. We have simulated the baseline operation scenario parameters by using SOLPS5.0 (B2.5-EIRENE) code package for a vertical lower single null (LSN) divertor configuration. The modeling shows that the peak heat load at divertor targets significantly exceeds the maximum engineering limit (i.e. 10 MW/m²) for the low density steady-state operations without any impurities puffing. The SOLPS simulations also demonstrate that Ar (or Ne, N2) puffing from the top of CFETR device is highly effective in mitigation of the divertor peak heat flux to below 10 MW/m², and both inner and outer divertor plates achieve detachment near the strike point with puffing rate reach a certain high level for the low density operation. In addition, the radiation loss fraction inside the separatrix will enhances and leads a reduction of the power flux across separatrix as impurities puff rate increase. Further simulations of different divertor geometries and configurations will be performed to optimize CFETR design.
In vessel electrical integration in ITER Tokamak

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ITER (Nuclear Facility INB-174) Vacuum Vessel is divided into 9 similar sectors where In-Vessel Diagnostics and Operational Instrumentation are located and which require the provision of Electrical Services. The electrical Services are connected through Feed-outs at the primary vacuum interface and distributed in the vacuum vessel by cable looms (up to 12 per sector). A cable tail will be routed from a junction box or from the component itself to link into the cable loom. The routing of the cable tails provide a considerable design challenge due to: 1) Number of the tails (>1500 cable tails) 2) Constraints to the in-vessel routing due to other in-vessel attachments and because of the need for shield blanket cut-outs 3) Assembly interactions between the tails and other in-vessel components 4) Finite loom capacity and in-port marshalling area space restrictions The manuscript will explain the engineering design of the routing process for each tail from junction box to feed-out and describe the solutions for the specific routing issues of each electrical component inside of the Vacuum Vessel. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.
P1.127

Manufacturing progress of first delivery sectors of ITER vacuum vessel thermal shield

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Thermal shield (TS) is one of the components in the ITER tokamak to minimize radiation heat load from vacuum vessel and cryostat to magnet structure that operates at 4.5 K. The TS main components (TSMC) are vacuum vessel TS (VVTS), cryostat TS (CTS) and support TS (STS). The TSMC are cooled by 80 K helium gas, which is supplied from the cryoplant via manifold pipes. The surface emissivity of the TSMC must be maintained under 0.05 by silver coating. ITER TS is being fabricated by the Korean supplier, SFA. The fabrication of all the nine 40 degree VVTS sectors are on-going since the start of material buffing in October 2014. Fabrication of VVTS proceeds according to the following processes: 1) material buffing, 2) plate cutting, 3) bending and forming, 4) 2nd buffing, 5) welding, 6) flange final machining, 7) pre-assembly of 40 degree sector, 8) final buffing, 9) silver coating, 10) final acceptance test. All VVTS shell segments are to be assembled by the flange joints, which are welded to the shells. Therefore tolerance requirement of the flange is strict for successful assembly of entire VVTS torus shape. Several pre-qualifications were performed prior to the manufacturing processes. All the welding joints were validated by non-destructive examinations and the inner surface of the welded cooling tube on The VVTS panel was inspected by a novel endoscope. Vacuum leak test was also performed in the test vacuum chamber. Currently, final machining of first delivery sectors (#6 and #5) was completed and the remaining works are pre-assembly of 40 degree sector and silver coating. Dimensional inspection method for the pre-assembled 40 degree VVTS sector and silver coating qualification method are briefly introduced in this paper. Lessons learned during the manufacturing of VVTS are also summarized.
Nuclear responses in the ITER IVVS port cell

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The ITER In-Vessel Viewing System (IVVS) consists of six identical units located at the B1 level of the Tokamak complex, at lower ports 3, 5, 9, 11, 15 and 17. They can be deployed to perform in-vessel inspections between plasma pulses or during a shutdown. When not in use, each unit is housed inside a dedicated port extending from the Vacuum Vessel (VV) outer wall to the port cell (PC), locked by a primary closure plate and equipped with a guide tube, shield blocks and a feedthrough for various services. In this work relevant nuclear quantities in the PC have been calculated by means of the MCNP-5 Monte Carlo code in a full 3-D geometry, including the IVVS and its shielding blocks geometry. A comprehensive MCNP model of the PC has been developed including a detailed description of the Bioshield plug, pipes, penetrations, cask rails and PC door. The neutron and gamma sources needed to perform the nuclear analyses have been defined taking into account both the contribution from the radiation streaming through the Lower Port and the gammas locally emitted by activated water circulating in the cooling pipes. Monte Carlo calculations have been performed to assess the radiation field inside the PC through neutrons and gamma maps. Absorbed dose during the ITER lifetime on sensitive components have been estimated in the PC area, in order to estimate the nuclear loads that the installed equipment have to withstand. Furthermore, the impact of the gamma-rays emitted by neutron-activated water circulating in the Primary Heat Transfer System have been evaluated on the PC environment: 3-D maps of the gamma flux, absorbed and biological dose rates during plasma operation are provided.
High spatial resolution heating for the ITER vacuum vessel with updated C-lite MCNP model

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Nuclear heating of the vacuum vessel (VV) is an important issue for the design and the safe operation of ITER. The heating distribution must be known with high accuracy to identify hot spots which may be crucial for the reliable operation. The VV is heated by neutrons passing through the blanket shield modules and gaps, and photons generated in the VV structure. The heating distribution is thus strongly affected by materials and geometry of in-vessel components. An accurate representation of these components is therefore a key prerequisite for reliable results of the nuclear heating distribution in the VV. To satisfy such requirements, C-lite, the reference model of ITER for neutron transport simulations with the Monte Carlo code MCNP, was updated with a semi-detailed representation of the in-vessel components (IVC) as currently designed for blanket rows 7 to 12. Semi-detailed IVC models in blanket rows 1 - 6 and 13 - 18 were already available, although corresponding to an earlier design stage. The engineering CAD models for blanket rows 7 to 12 were processed according to the needs of neutronics simulations, converted into MCNP geometry representation and then integrated into C-lite, together with the available models of blanket rows 1-6 and 13-18. The updated C-lite model was applied to compute distributions of the nuclear heating in the VV with the MCNP6 Monte Carlo code using mesh tallies with a resolution of 2 cm. The calculations were performed on the HELIOS supercomputer located in Rokkasho, Japan. The paper describes the updating of C-lite in detail and presents the results of heating calculations using both analogue Monte Carlo and various variance reduction techniques. The new results confirm the VV hot spots obtained previously for C-lite with a simplified IVC representation but give less conservative results for VV regions behind the updated IVCs.
Influence of welding on stiffened box structure during fabrication of ITER VV Inboard Segment

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ITER vacuum vessel (VV) is composed of 9 sectors, and each sector is completed through an assembly of 4 segments which are independently fabricated. Compared with Upper, Equatorial and Lower segment which have relatively large curvature in a 3 dimensional configuration, Inboard segment is the most difficult in aspect of a welding distortion control although it seems to be simply in fabrication due to relatively small change of curvature. The mock-up of Inboard segment had been fabricated with a 3 m in a length and 40 degree in a width. At that time, even though the 34 mm of welding distortion was anticipated as a result of the analyses, it could be practically reduced up to 14.5 mm by the help of the special welding fixture developed to prevent welding distortion. However, a welding distortion of a real Inboard segment is expected to be increased more than 2 times compared with the mock-up because its length is about 7 m which is longer than the mock-up. Therefore, it is very important to accurately measure a welding distortion according to a manufacturing sequence, and take a feedback control of a welding distortion in subsequent welding operation. In this paper, the effect of a major welding operation on a welding distortion is evaluated for the fabrication of Inboard segment. For this evaluation, the 3 dimensional variations in the measurements of welding distortion of Inner shell were recorded after the completion of welding of 18 keys, 48 flexible support housings, divertor support, and whole length of 3 poloidal ribs. These results will be used to design welding fixtures for the assembly of In-wall shield and Outer shell welding in order to control welding distortion within 20 mm which is an allowable requirement from ITER.
P1.131

Design, Validation and Manufacturing of ITER Vacuum Components, & Leak Localization

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The ITER vacuum system will be one of the largest, most complex vacuum systems ever to be built and includes a number of large volume systems such as the Cryostat (~ 8500 m\(^3\)), Torus (~1330 m\(^3\)), and the Neutral Beams (~180 m\(^3\) each). The vacuum system comprises of custom and commercially available components and adapted commercial vacuum technology. For a component with a nuclear safety function validation of the design is required. Validation may take different forms including by analysis and/or test. Additionally the manufacturing must be controlled to ensure the equipment conforms to the validated design. Where similar components are used a program is in place to standardize these components to a common design. Leaks are expected to account for a loss of operational availability if timely localisation and repair cannot be performed. The design of the ITER in-vessel systems are such that localisation of a leak must be performed with sub-centimeter resolution such that the risk of removing a leak tight component in error is reduced. Due the progression to an active environment, traditional methods of leak localization may not be applicable. Hence a challenge is to develop methods of leak localization capable of operation in the ITER environment, with a minimum of human intervention. In this paper an overview is given of the nuclear safety functions of the vacuum system and different routes to achieving validated functions provided for a variety of components. Experience gained from the design, validation and manufacturing process is described. A concept for the localization of water leaks based on the spectroscopic detection of plasma excited hydroxyl (-OH) is described. Also, details of a methodology, based on the pressurisation / depressurisation of water feeds, for the localisation of leaks from the in-vessel blanket system is provided herein.
Reliability analysis of the ITER LCTS double plate support

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ITER Thermal shield (TS) is a thermal barrier in the ITER tokamak to minimize heat load transferred by thermal radiation from the hot components to the superconducting magnets operating at 4.2K. TS supports are designed to endure a dead weight, seismic load, electromagnetic load and thermal loads. In the design and analysis of the TS supports, deterministic values of the geometry or dimension of the components, load condition and material properties are used. However, the actual dimensions will be random during the manufacturing process and controlled in the form of tolerance. Mechanical properties of the material are varied due to uncertainties in the chemistry and fabrication procedures used. The loads acting on the support also have variability due to fluctuation of specific weight of the material as well as changes in the dynamic load such as seismic load and electromagnetic load. In the deterministic design practice, the safety factors are used to account for the variability of the system. On the other hand, probability of the failure will be specified after reliability analysis under the probability distribution of the input parameters. In this paper, design sensitivity study will be carried to see the effect of the variation of the geometry, material properties and load conditions. And reliability analysis will be performed and probability of the failure will be verified on the CTS support. Dimension, material properties and load condition are to be treated as random processes for the reliability analysis of the support.
Specific design and structural issues of single crystalline first mirrors for diagnostics

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The first mirrors of ITER diagnostic systems are the most vulnerable ones since they are directed to the plasma and are subjected to erosion and intensive impurity deposition. In order to prolong the lifetime of the first mirror and to keep its high optical performance and maintainability, single crystalline molybdenum and rhodium have been considered as mirror materials, subject to intensive investigations including R&Ds and mirror cleaning studies. The paper presents specific design and structural issues of the first mirror considered for the ITER core charge exchange recombination spectroscopy (cCXRS) several years ago [1, 2] where a mirror size reaches ~ 300 mm. Such large mirrors can be assumed as a generic ones for a solid middle mirrors (up to 100 mm in diameter) and for composed large mirrors (diameter > 150 mm) that are potentially made of an assembly of smaller pieces. The good design for the single crystalline mirror shall provide: - a reliable structural connection between mirror assembled pieces and a substrate; - acceptable mirror temperatures and thermal distortions providing the functionality of the optical system; - cleaning suitability; - mirror positioning stability within the required limits during operational and dwell time; - an acceptable level of mechanical and thermo-mechanical stresses; The mirror design is supported by extensive structural multi-physical analysis. To prove the feasibility of the proposals, a wide spectrum of technological R&D efforts is required.
P1.134

ESPvN regulation applied to the Port Plug Structure for ITER Diagnostic system

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ITER Diagnostic Port Plugs will operate with water at high pressures and temperatures. Because of these conditions of operation, the diagnostic Port Plugs are under the French Regulation on Pressure Equipment / Nuclear Pressure Equipment. This paper focuses on the assessments performed in order to substantiate application of Article 2 paragraph II of French decree 99-1046 relieving diagnostic port plugs structures from provisions appearing in Title II and III from above mentioned Decree. Being close to the plasma, diagnostic Port Plug structures and Diagnostic First Walls (DFW) contain water for cooling during operation. Water is also used for heating during bake-out. Heat extraction studies demonstrate the need to use water pressurized at up to 48 bars and 250 degree. These structures are designated as “Pressure devices” and, therefore, need to follow French regulation. This paper describes the design of the Port Plug Structure and DFW from a point of view of applicability of Article 2 paragraph II of French decree 99-1046 relieving diagnostic port plugs structures from provisions appearing in Title II and III from above mentioned Decree. Analysis of key loads will be presented to justify the conclusions. Optimisation of the design to ensure a fully safe system over the life cycle will be discussed.
Ultra-sensitive ultrasonic testing method on bi-metallic explosion plate for ITER low-temperature superconducting joint box

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The cryogenic superconducting joint box is an important part of ITER HTS current leads, which is made of Copper-316L bi-metallic explosion bonded plate. The bimetal interface of the joint has the direct effect on the mechanical properties of the joint and testing performance at low temperature. This paper describes work on the development of water immersion ultrasonic testing technology, and its application on bi-metallic explosion bonded plate. It also gives 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 was 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 > 2mm² in the final box should be confirmed.
Thermo-mechanical analysis of the Tore Supra WEST Cooling Water System

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The French Tore Supra tokamak is upgraded in an x-point divertor fusion device in the frame of the WEST (W-for tungsten-Environment in Steady-state Tokamak) project, launched in support to the ITER tungsten divertor strategy. The WEST project aims to test actively cooled tungsten Plasma Facing Units (PFU) under long plasma discharge. As the existing cooling loop B30 cannot ensure the cooling of the W divertor elements under ITER nominal conditions, a new pressurized water loop must be designed to exhaust all the heat coming from the plasma and transmitted to the in-vessel components. It includes additional piping networks of around 50m³ connected to the existing loop in order to cool components such as stainless steel vessel protection panels, upper and lower divertor copper coils, baffle, bumpers and ripple protection. During plasma operation, the water inlet temperature is 70 °C and the pressure at the inlet of the pump is 2.4 MPa while baking is performed at 200 °C and 2MPa to achieve the outgassing of components inside the vacuum vessel. As the pressure drop of the divertor coils is higher than the pressure drop of the other PFCs, the cooling of these coils is ensured by an independent water loop with a new centrifugal pump. This paper presents the thermo-mechanical analysis of this system including piping stress and supports analysis according to CODETI standards.
P1.137

Qualification, manufacturing and assembly of the WEST divertor structure and coils

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In order to fully validate “ITER-like” actively water cooled tungsten plasma facing units, addressing the issues of long plasma discharges, an axisymmetric divertor structure has been studied and manufactured for the implementation in the WEST (Tungsten (W) Environment in Steady state Tokamak) tokamak platform. This assembly, called divertor structure and coils (4m diameter, 20 tonnes), is composed of two stainless steel casings containing an actively water cooled (up to 180°C, 4MPa) copper winding pack designed for a current in the range of 12.5kA (up to 1000s). It must sustain harsh environmental conditions in terms of ultra-high vacuum, high temperatures and electrodynamic loads. One major difficulty is the induction brazing assembly of individual bended conductor portion inside the vacuum vessel and the consecutive sealing of the casings by TIG welding. Therefore development activities have been carried out on a scale one dummy coil, such as brazing, assembly, thermal cycling and electrical insulation tests (5kV ground voltage). Whereas the brazing assembly technics and the conductor installation were validated without major difficulties, different technical solutions for the electrical insulation had to be tested. The chosen solution is a resin epoxy impregnated fiber glass fabric layered around the conductors followed by a polymerization procedure. In parallel the manufacturing of divertor structure components started in the second half of 2013 with a total delivery at the end of 2015. The paper will illustrate the technical developments which have been performed in order to fully validate the design. It concerns mainly the dummy coil and the complex conductor installation procedure assisted by virtual reality tools. The manufacturing methods proposed by the contractor in order to fulfill the technical requirements will be also addressed. Finally the processes and associated tools used in order to implement this large component inside the WEST vacuum vessel will be detailed.
Manufacturing and Installation of the JT60-SA Helium Storage Vessels for the Cryogenic Plant

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The JT-60SA Tokamak is provided with a cryogenic system with a refrigeration capacity of 9KW (equiv.) at 4.5 K. Before commissioning and during occasional warm-up periods the total 3.6 t helium inventory is stored in six pressure vessels, which have been procured by Europe. Each vessel is 22 m long, has a diameter of 4 m, a 250 m³ volume, and weighs about 73 t. As the vessels will store pure helium, the tightness and cleanliness requirements were quite demanding. One of the vessels is also used to receive the cold helium (20K) from the cryogenic system quench line, following a fast discharge of the superconducting coils. A special 18 m long helium diffuser system and a thermal barrier connector at the quench line flange has been designed and manufactured to avoid local chilling of the vessel wall below the minimum allowed temperature of the material, due to the ingress of the cold helium. The detailed design, manufacturing and testing of these large components have been completed on time and budget. The vessels have been shipped to Japan and transported from the port of entry to the Naka research centre. The vessels have then been installed on their foundations using a 500 t crane using a detailed assembly procedure. This paper reports the design, manufacturing, testing and installation of the six large helium storage vessels for the JT60-SA cryogenic system.
Three-dimensional condensation regime of steam injected into water at sub-atmospheric conditions


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Important challenges for fusion technology deal with the design of safety systems designed to protect the Vacuum Vessel (VV) in the case of pressurizing accidents like the LOCA (Loss Of Coolant Accident). This accident is caused by the failure of a number of elements of the Tokamak Water Cooling System and may result in relevant consequences for the integrity of the reactor. To prevent or to mitigate structural damages, the solution proposed is a safety system able to quick condense the released steam feeding it into water at sub-atmospheric conditions. This system, connecting the VV to an auxiliary Pressure Suppression Water Tank through a relief line, is equipped with a rupture disc allowing to discharge the released steam. In this framework, an important role is played by the Direct Contact Condensation of steam in water, which is the investigated process to be used to lower the sudden overpressure within the VV. The DCC of steam at atmospheric pressure has been extensively analyzed and experimentally investigated in the past decades for BWR design optimization purposes. Nevertheless, up to date there are no explicit experimental data available for sub-atmospheric pressure conditions. The originality of this study relies on the experimental work that has been done at the DICI of the University of Pisa to provide extended experimental data, necessary to allow a better assessment of DCC of steam phenomena. To analyze the steam condensation regime, the pressure was ranged between 4.2 and 120 kPa while the water pool temperature from 10 up to 85°C. About 300 condensation tests were performed allowing investigation of the combined influence of steam mass flux, water temperature and pool pressure on the steam condensation phenomenon. Procedures adopted and results are duly presented and discussed, focusing upon the efficiency of the steam condensation for all examined conditions.
The flexible in-vessel inspection system (FIVIS) for EAST is a unique 10-degree-of-freedom manipulator for its serial structure of arcuate deployed Big Arm and its planar Small Arm (end effector): the Big Arm takes the Small Arm to all positions of the toroidal vacuum vessel (VV) along its equatorial plane, achieving a full coverage of VV’s first wall. In the in-vessel inspection process, the Big Arm will stretch out as a hanging beam, the full extended configuration of which will reach a half of the toroidal VV. Significant offset from the center of D-shape section of VV is observed at the full extended posture of Big Arm due to its gravity and external payload during the prototype operation test, making it difficult to maintain the position accuracy of planned trajectory for inspection process. In this paper, By structural analysis and the finite element analysis methods, a simplified rigid-flexible model of FIVIS is adopted to calculate the deformations for fully extended and fully extracted configurations of Big Arm and effectors. The position accuracy tests have been performed by laser tracking system to validate the effectiveness of proposed stiffness model. A deflection compensation method has been presented to improve the stiffness of Big Arm. Future work show that modal analysis methods is an effective tool to improve the dynamic performance of high precision position control of FIVIS.
Design of a Standalone Joint Module toward Real EAST In-vessel Operation Use

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The remote handling in-vessel inspection manipulator specially developed for EAST superconducting tokamak has proven its kinematics feasibility in scale one toroidal vessel and its survivability under 120 °C high temperature. To adapt this manipulator for real in-vessel operation, most of its joint components, such as motors and reducers, must be isolated in sealed spaces to prevent possible contamination to the in-vessel vacuum ambient. In order to verify the feasibility of combining vacuum sealing techniques with the previous in-vessel inspection manipulator solution, we have designed a standalone rotary joint module as prototype for relevant experimental tests. The joint module has a standard stainless bellow tube to seal all its ordinary commercial joint components from the outside, while servo control wires, sensor wires and active cooling water tubes inside the sealed chamber are led out by special vacuum feedthroughs. Basic motion tests have been carried out under equivalent in-vessel environment, i.e. 120 °C high temperature and 10e-5 Pa vacuum. This paper mainly presents the vacuum sealing design and related test results of this prototype joint module. A comparison with previous in-vessel operation manipulator solutions, such as AIA from CEA and IVVS for current ITER program, is also involved. Further work will concentrate on adapting the whole in-vessel manipulator for vacuum environment operation based on this preliminary design experience of bellow tube sealed joint module.
Vibration suppression control of EAMA/EAST Articulated Maintenance Arm

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EAMA (EAST Articulated Maintenance Arm) is an articulated serial robot arm working in experimental advanced superconductor tokamak for inspection and maintenance. Redundant flexible structure of EAMA increases reach capability, however, it reduces accuracy and speed due to the compliance introduced into each joint. This deteriorates EAMA into oscillation and produces undesirable disturbance. In this paper a nonlinear model predictive controller with acceleration feedback is designed for active vibration damping. The flexible joints dynamic model of EAMA is established. The proposed cost function, constraints and estimated prediction horizon guarantee closed-loop stability. The control method reduces the oscillation amplitude efficiently by minimizing kinetic energy loss. The experimental result shows that EAMA obtained closed-loop performance is improved when compared to a PD controller.
P1.144

**Error compensation strategy of EAST articulated maintenance arm robot based on static stiffness modeling**

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EAST Articulated Maintenance Arm (EAMA) is a highly redundant serial robot system with 11 degree of freedoms (DOFs) in total. It will allow remote inspection and simple repair of plasma facing components (PFCs) in EAST vacuum vessel (VV) without breaking down the ultra-high vacuum condition during physical experiments. Due to its long-reach mechanisms with a weight more than 100 kg, the gravity effect will cause huge flexible deformation, which is unacceptable for running inside a narrow and complex-shaped space as EAST VV. To solve this problem, a mathematic model for the static system stiffness of flexible robot segments has been built in this paper by utilizing the Matrix Structural Analysis (MSA) method, in which case, deformation prediction in arbitrary positions and postures can be obtained by solving the stiffness matrix equations. Furthermore, an error compensation strategy based on the deformation prediction results has been developed to improve the motion accuracy of whole robot systems. Keywords: EAMA robot; Static stiffness modeling; Deformation prediction; Error compensation
Design status of DEMO blanket primary heat transfer system

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The design requirements for the DEMO Blanket Primary Heat Transfer System, both for the water and helium concepts have been defined. The plasma facing components cooling circuits have to fulfill several requirements dictated by safety and operational criteria. In particular, the Blanket PHTS of a fusion reactor shall transfer the heat load coming from the plasma to the secondary side to allow power removal from the in-vessel components and produce high quality steam to be sent to the power conversion complex. The design solutions should meet the different characteristics of the various blanket concept and the pulsed operation of the reactor. The interactions of high energetic neutrons with the Blanket internals lead to the generation of Activated Corrosion Products, Tritium and Nitrogen radioisotopes. These fusion-specific radiological products shall be safely confined by the PHTS during the various machine states to make as minimum as achievable the radiological doses to the workers, people and environment.

The paper will outline the PHTS design requirements as well the preliminary design for both water and helium blanket cooling systems in terms of circuit layout, located inside the tokamak building, size of piping and equipment, thermodynamic efficiency and safety characteristics.
Europe is developing two reference tritium Breeder Blankets concepts that will be tested in ITER under the form of Test Blanket Modules (TBMs): i) Helium-Cooled Lithium-Lead (HCLL) which uses liquid Pb-16Li as both breeder and neutron multiplier, ii) Helium-Cooled Pebble-Bed (HCPB) with lithiated ceramic pebbles as breeder and beryllium pebbles as neutron multiplier. Both concepts are using the EUROFER97 steel as a structural material and pressurized Helium technology for heat extraction (8 MPa, 300-500°C). First part of the paper reviews the progress achieved in the conceptual design development of the both concepts, namely, the HCLL and HCPB TBM-sets comprising of TBMs itself, the associated shield and mechanical attachment providing the required structural connection between the TBM and shield. Second part of the paper overviews activities focused on qualification of EUROFER97 structural material, introduced under a probationary phase in the nuclear components design and construction code RCC-MRx, and identification/analyses of gaps in the respective material database to be filled in. Additionally the available design rules in the code are reviewed to verify their applicability to the specificities of EUROFER97 steel and to TBM design and fabrication. Third part of the paper overviews the current development of fabrication technologies and procedures applied for manufacturing of TBM box assembly and TBM sub-components, like, HCLL and HCPB cooling plates, stiffening plates, and first wall and side caps. The used technologies are based on fusion and diffusion welding techniques taking into account specificities of EUROFER-97 steel. Fourth part of the paper presents on-going activities towards qualification of functional materials for TBM applications, namely, Li ceramic breeder, beryllium multiplier and Pb-16Li eutectic alloy. Preparation of the functional materials’ Material Assessment Reports is discussed. In particular, results obtained in post-irradiation examination of ceramic breeder and Be pebbles irradiated at HICU and HIDOBE campaigns are presented.
Status of the platform for integration and maintenance tests of the European TBM systems

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The Test Blanket Module (TBM) and its associated ancillary systems (including cooling systems, tritium extraction system, coolant purification, PbLi loop, I&C) form the Test Blanket System (TBS). The TBSs will be fully integrated in the ITER machine and buildings. Therefore, testing of the TBS integration and maintenance in ITER port cell prior to its installation and operation in the ITER machine is one of the keys points to ensure success of the TBM program. A feasibility of all the planned TBS maintenance, installation and inspection operations in the port cell should be proven first, which will require also their demonstration at dedicated large scale mock-ups. A full scale test platform reproducing the ITER equatorial port cell #16, corresponding TBS interfaces and the main components was designed and manufactured in Centrum výzkumu Řež (CVR). The platform was designed in a way to offer testing, optimization and validation of the TBS related maintenance tools, operations and procedures to be performed in the port cell. Moreover, it allows also testing of accessibility conditions to various interfaces and components inside the port cell. Although it reproduces the TBS interfaces and the main equipment corresponding to the European TBS configuration, its modular design makes it possible to adapt them so that other, non-EU TBS configurations could be tested at the platform as well. The paper describes the current status of a project of the full scale test platform for integration and maintenance tests of the European TBM systems in ITER port cell #16. It contains also description of its design, parameters, characteristics and the first operational experience.
Europe is developing two reference tritium breeder blankets concepts that will be tested in ITER under form of Test Blanket Systems (TBS): (i) the helium-cooled lithium-lead (HCLL) which uses liquid PbLi as both breeder and neutron multiplier, (ii) the helium-cooled pebble-bed (HCPB) with lithiated ceramic pebbles as breeder and beryllium pebbles as neutron multiplier. One of core documents to be prepared in view of satisfactory licensing of each of European HCLL & HCPB TBS systems is their ‘Maintenance Plan’. This document, which becomes fundamental for ensuring sound performance and safety of TBS along ITER operational phase, shall include information as relevant as maintenance organization, preventive and corrective maintenance task procedures, condition monitoring for key components, or spare parts plan, just to enumerate some of them. In compliance with ITER Maintenance policies, first steps aimed to define nuclear maintenance strategy for key HCLL & HCPB TBS components are been conducted by F4E in collaboration with industry, being purpose of such preliminary nuclear maintenance studies: - identification of requirements and constraints for maintenance activities; - preliminary assessment of maintenance tasks sequence required for key components of each sub-system: Helium Cooling System (HCS), Coolant Purification System (CPS), Tritium Extraction System (TES) and PbLi Loop; - preliminary assessment of necessary Maintenance tools and equipment; - preliminary estimation of maintenance task duration. The mentioned maintenance studies are based on TBS conceptual design and are intended to challenge their design from maintenance perspective so as to take on board at early design phase possible design recommendations that could facilitate maintenance activities particularly on those areas for which reduction of occupational radiological exposure following ALARA principle is strictly required. After brief recall of design features of HCLL & HCPB TBS, above mentioned topics are discussed in this paper, paying attention to main conclusions in terms of design recommendations.
Myth of initial loading tritium: modelling DEMO fuel system in power ascension tests

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It is widely believed that fusion DEMO reactor will need significant amount of tritium at the beginning of its operation. However, the authors have pointed out that steady deuterium operation can produce sufficient tritium in a reasonable period of DD operation by DD reaction followed by exponential breeding in the blanket. The present study further suggests that realistic Power Ascension Tests (PAT) of DEMO can produce its tritium to be needed in the series of tests by its own program until reaching steady state full power operation, and thus no additional supply is needed. Closed tritium fuel plant was described by a system dynamics model, and analyzed considering realistic PATs of DEMO, that will be mainly pulsed DD and low concentration DT. Primary fuel cycle is composed of plasma exhaust evacuation, isotope separation by cryogenic distillation, storage and blanket tritium recovery. Secondary systems such as tritium recovery from water and solid waste, secondary confinement to capture permeated and leaked tritium is also analyzed to recycle tritium with longer time constants. Although no actual PAT plan for fusion DEMO is available, previous PATs for new fission reactors provides realistic scenarios. Typical PATs require years of operation from zero power criticality to full power, with pulsed power output and long dwell time between them. Output power is gradually increased in PATs to check the functions of reactor systems and components. In the case of fusion DEMO, zero power criticality corresponds to DD operation. While plasma may be fired in pulses, tritium plant is continuously operated to recover all the tritium produced by the DD and low DT burn. Depending on the different time constant of tritium retention in components, tritium is transferred by deuterium purge, and high concentration tritium is finally collected in the storage, to be available for the next tests.
The basis of a thermonuclear fusion reactor is neutron source (FNS) based on the tokamak [1]. FNS should provide steady flow of fusion neutrons with a capacity of 10-50 MW, which reached close to the pulse values of existing installations JET and JT-60U. Fuel cycle technologies (FC) is one of the key elements for the FNS. FC systems should provide treatment and storage of deuterium and tritium, as well as the processing of the fuel mix in all systems of a thermonuclear reactor. These technologies have to be developed significantly, because the technical solutions chosen ITER project can be used in FNS is only partially due to steady state operation of the plant, the higher neutron fluxes and fluxes of tritium fuel cycle elements. To assess the distribution of tritium in fusion reactor systems and components “tritium plant” is necessary to carry out a dynamic simulation of all system elements allowing for the operation of the tokamak. Such calculations are now performed using the code «FC-FNS» [2]. The code allows the calculation of tritium flows and stocks in tokamak fusion systems. To close the FC processes of tritium in the hybrid blanket was considered. The report is a conceptual diagram of a stationary fuel cycle FNS with 3-50 MW of fusion power, given current estimates of the distribution of tritium in fusion reactor systems and components “tritium plant.” Calculations of tritium flows and accumulation have been carried out for two different cases of the fuel mixture for neutral beam injection (NBI) system. [1]. B.V. Kuteev, at al. // Published 26 June 2015 © 2015 IAEA, Vienna Nuclear Fusion, Volume 55, Number 7. [2]. Anan’ev S.S. et al. Concept of DT fuel cycle for a fusion neutron source // Fusion science and technology vol. 67 mar. 2015
An equation of state for lead lithium eutectic

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Thermal hydraulic and accident analysis codes such as RELAP5-3D and MELCOR rely on an equation of state to specify all the thermodynamic properties of fusion-relevant working fluids such as PbLi. The existing liquid metal fluid properties in both RELAP5-3D and MELCOR are based on a five parameter “soft sphere” equation of state for which parameter sets that approximately reproduce experiment data were available in the literature. The PbLi parameters were based on a linear, mass-weighted average of the parameters for pure Pb and pure Li, which does not always result in adequate agreement between the equation and experimental data for PbLi, much of which was not available at the time the original equations were implemented. In order to address this shortcoming, we describe here a modification and non-linear least squares fitting of the parameters to closely match available experiment data for the density, sound speed, and specific heat capacity of liquid PbLi over a range of temperatures. The new equation has been implemented in both RELAP5-3D and MELCOR, and we make comparisons with both codes to experiment data to verify the implementation. Transport properties such as the thermal conductivity and viscosity, though not derived from the equation of state, have also been updated.
A new HCPB breeding blanket for the EU DEMO: evolution, rationale and preliminary performances

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The Helium Cooled Pebble Bed (HCPB) Breeding Blanket (BB) is one of the four BB concepts being investigated in the EU for their possible implementation in DEMO. During 2011-2013 initial HCPB BB conceptual studies were performed based on a design extrapolation from the ITER’s HCPB Test Blanket Module, leading to the so called “beer-box” BB concept. During 2014 the “beer-box” BB concept suffered several design changes so as to meet different counteracting nuclear, thermo-hydraulic and thermo-mechanical requirements, specially evidencing that the concept was not flexible enough to meet the tight TBR requirements (i.e. TBR ≥ 1.10). Additionally, the complex manifold system with unbalanced helium mass flow needs in each of the two redundant cooling loops made the concept complex. However, parametric studies during 2015 revealed that the HCPB concept have potential for larger nuclear performance, as well as potential for a significant simplification of the cooling internals by redefining the cooling plates and the architecture of this blanket, making the flow scheme symmetric. This paper describes the new HCPB blanket concept based on a “sandwich” structure of cooling plates with integrated helium manifold systems. The former complex manifold backplate system has been compacted and integrated in the cooling plates, releasing about 300mm of radial space that can be used now for increased tritium breeding, shielding or reinforcement of the Back Supporting Structure (BSS). Detailed neutronic analyses confirm a TBR of at least 1.20. Preliminary analyses show good thermo-hydraulic behaviors of the concept and preliminary thermo-mechanical analyses also indicates that the design should be able to withstand off-normal in-box LOCA scenarios up to a level C according to the RCC-MRx code. Future optimization activities are described, which shall lead to a concept meeting all the BB requirements with still some margin for mitigation, in the case of future changes in the tokamak configuration.
Neutronic analyses for the optimization of the advanced HCPB breeder blanket design for DEMO
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Within the Power Plant Physics and Technology (PPPT) programme of EUROfusion, a major development effort is devoted to the conceptual design of a DEMO reactor which has the capability to breed Tritium for self-sufficiency. This DEMO is assumed to be suitable for the accommodation of any blanket type out of the existing concepts. For the neutronics analyses, a generic DEMO model is thus set-up which serves as common basis for the integration of blankets of the considered four concepts. The generic geometry model, based on a CAD model, is then refined for particle transport simulations using the MCNP code. The generic CAD model with voided blanket space is used for the arrangement of the HCPB blanket boxes without interior structure around the plasma. The configuration of each module is adjusted to the specific position in the reactor and follows the given first wall contour. This model is converted to a MCNP geometry model by making use of the CAD to MCNP geometry conversion tool MCad. The empty space of the blanket modules is then filled with the internal structures representing a detailed HCPB design by making use of the repeated structure feature of the MCNP code. This work gives an overview of the neutronic analyses to support and optimize the advanced HCPB blanket concept for DEMO. Full scale 3D Monte Carlo particle transport simulations were performed to this end with the MCNP5 code employing a very detailed HCPB DEMO torus sector model. Optimization of the tritium breeding performance was performed to determine the blanket configuration satisfying tritium self-sufficiency requirement. Different geometry configurations and material arrangements suggested by safety considerations were investigated and implemented in the final blanket design. Various integral and differential nuclear responses necessary for the reactor design and safety analyses were provided on 2D and 3D maps.
Lithium density and tritium release behaviour are key properties in the design and synthesis of Li-containing solid breeders for the helium cooled pebble blanket (HCBP) concept. Radiation and high temperature may give rise to changes in both material composition and microstructure, hence important aspects including chemical compatibility and tritium production/extraction effectiveness may be strongly affected during reactor operation. Experimental validation of these materials must contemplate examination under relevant conditions. The Radiation Induced Permeation and Release (RIPER) facility, in the beam line of the 2 MV Van de Graaff electron accelerator at CIEMAT has been conceived as a reference laboratory to measure adsorption, absorption, desorption, and permeation, as well as decomposition-vaporization during irradiation at variable temperatures and ionizing dose rates, under different environments (vacuum, and He and H isotopes at different pressures). Radioactive tritium handling issues are avoided by extrapolating tritium data from H2 and D2 results. With this system lithium loss, driven by radiolysis at ~ 400 Gy/s and thermal vaporization from 20 to 500 °C, has been monitored for pellet and pebble-shaped samples of lithium orthosilicate (Li4SiO4) and lithium metatitanate (Li2TiO3) composites produced at CIEMAT, using a PrismaPlus QMG 220 (Pfeiffer) mass spectrometer. Surface composition has been additionally examined by means of XPS to identify possible changes.
Deuterium thermally induced desorption for HCPB breeder compositions during electron irradiation at relevant temperatures

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The tritium release behaviour of candidate ceramic materials for the HCPB breeder concept is still an issue. High experimental costs, long experimental periods, and handling difficulties for activated materials after being tested in experimental fission reactors have motivated the validation of alternative methods for testing the gas desorption behaviour of tritium breeder materials. In the framework of EUROFusion, desorption experiments are being performed during irradiation by means of accelerated electron and ion beams to simulate the electronic and structural changes induced in breeder ceramics due to neutron radiation. The aim of this four year study is two fold: an understanding of the diffusion and release phenomena which take place in a ceramic pebble bed, and the efficiency of these radiation sources will be validated for a DEMO-like simulated environment. Deuterium desorption results for advanced lithium orthosilicate pebbles, mechanically improved by additions of lithium metatitanate (developed at KIT) are presented here. Pebble beds of different composition were first deuterium loaded under ionizing radiation and breeder temperature operational conditions, and then thermally induced gas desorption recorded during heating. The experimental results indicate that the gas release is more efficient when the pebble bed is loaded at temperature and under irradiation. Furthermore the observed enhancement of deuterium desorption efficiency has been attributed not only to the ionizing radiation, but also to the increase of surface roughness, which changes with the addition of the reinforcing second phase, lithium metatitanate. Fusion operational conditions in advanced breeder blanket ceramics will then facilitate the release of the tritium produced due to lithium transmutation reaction.
DNS of Turbulent Heat Transfer in pipe flow via MPI+OpenMP for multi-CPU

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The simulation plays an important role to estimate characteristics of cooling in a blanket for such high heating plasma in ITER-BA. An objective of this study is to perform large -scale direct numerical simulation (DNS) on heat transfer of turbulent flow on coolant materials assumed gas flow. The coolant flow conditions in ITER-BA are assumed to be Reynolds number of a higher order. To investigate the effect of Reynolds number on the scalar structures, the Reynolds number based on a friction velocity and a pipe radius was set to be $Re_τ = 2100$. The number of computational grids in 512 nodes is $4096 \times 1024 \times 1536$ in the $z$-, $r$- and $φ$-directions, respectively. The detailed turbulent quantities such as the mean flow, turbulent stresses, turbulent kinetic energy budget, and the turbulent statistics were obtained.
Effective thermal conductivity of advanced breeder pebble beds
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All solid breeder concepts, considered to be tested in ITER, make use of lithium-based ceramics in the form of pebble-packed beds as tritium breeder. A thorough understanding of the effective thermal conductivity of the ceramic breeding pebble beds in fusion relevant conditions is essential for the design of the breeder blanket modules of the future fusion reactors. An experimental set-up for the investigation of the effective thermal conductivity of ceramic pebble beds was designed and assembled. The hot wire method was selected to measure the thermal conductivity of ceramic pebble beds. Measurements of the effective thermal conductivity of polydispersed lithium orthosilicate pebble beds with different lithium metatitanate contents were performed. The EU reference tritium breeding material was investigated as well. The effective thermal conductivity was investigated in the temperature range between RT and 600 °C. Experiments were performed in helium atmosphere in the pressure range 0.12-0.4 MPa with a compressive load up to 6 MPa. The initial packing factor of the beds was approx. 64 %. The results show no significant influence of the chemical composition of the solid material on the bed’s effective thermal conductivity. An increase of the effective thermal conductivity with the temperature was observed for all investigated compositions. The results show a slight increase of the effective thermal conductivity with the applied load. The halving of the helium pressure results in a slight reduction of the effective thermal conductivity at all investigated temperatures. However, the reduction of the helium pressure from 0.4 to 0.12 MPa leads to a significant reduction of the bed’s effective thermal conductivity.
Experimental study on effective thermal conductivity of pebble beds for fusion blankets

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Solid blanket is a core candidate of blanket structure for CFETR (Chinese Fusion Engineering Testing Reactor), and the effective thermal conductivity of ceramic pebble beds is a very significant parameter for the thermo-mechanical design of solid blankets. In order to obtain the effective thermal conductivity, theoretical calculation and experimental measurement are two common methods. Compared with theoretical calculation, experiment measurement is more accurate. An experiment platform was designed by University of Science and Technology of China (USTC). The platform using transient thermal probe with Monte Carlo inversion method which can improve probe precision. The singer size pebbles were contained in the stainless steel tube with an outer size diameter of 48mm, the length of 450mm, and the tube was put in the tube furnace. Singer size Li₄SiO₄ or Ti₂TiO₃ pebbles with the diameter of 1mm, and the packing factor of 64% were tested in the experiment. The experiment temperature ranged between room temperature to 800 °C, and helium pressure ranged from 0.1 MPa to 0.3MPa with very low velocity. The results presented in this work will help to create a database of the effective thermal conductivity of Li₄SiO₄ and Li₂TiO₃ pebble beds. Keywords: Thermal conductivity; Pebble beds; Probe; Monte Carlo inversion method
Theoretical and Experimental Study on Effective Thermal Conductivity of Pebble Beds for Fusion Blanket

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Tritium breeder pebble bed plays a vital role in tritium breeding for fusion solid blanket. And thermo-physical properties of it affect the thermo-mechanical and structural design of solid blanket directly. Theoretical and experimental study on effective thermal conductivity of ceramic pebble beds have been carried out in this paper. Firstly, a new theoretical model, coupling the contact areas with bed strains, was developed to predict the effective thermal conductivity of mono-sized ceramic pebble beds. The influences of parameters such as properties of pebble and gas materials, bed porosity, pebble size, gas flow, contact area, thermal radiation, contact resistance, etc. were all taken into account in this model. Experimental platforms also have been built to take a measurement of effective thermal conductivity of ceramic breeder pebble beds (e.g. Li4SiO4 and Li2TiO3 pebble beds). Two experimental platforms using transient thermal probe method and transient plane source method respectively were successfully under operation. Li4SiO4 and Li2TiO3 pebble beds with 1 mm diameter and temperature window from 100°oC to 800°oC were considered in the experiments, and the helium purge gas with 0.1˜0.3MPa were studied to assess the influence of purge gas pressure on effective thermal conductivity of pebble bed. Keywords: effective thermal conductivity, pebble beds, theoretical method, experimental platform, solid blanket
The thermal analysis to pebble bed of CFETR

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Thermal transport efficiency of a tritium breeding pebble bed can strongly affect tritium self-sufficiency of the magnetic confinement fusion solid breeding blanket system. The effective thermal conductivity of the pebble bed is related not only to its configuration, such as dimensions, pebble size, and pebble material porosity, but also to its environment, such as helium temperature, flow velocity, and purge throughput. Currently under development, can analyze the thermal mechanical properties of individual pebbles of a pebble system, accounting for the effects of pebble fracture. Combining Computational Fluid Dynamics (CFD) with DEM enables analysis of the complex interactions between the solid and gaseous phases. This proposal aims to carry out numerical simulation of a bed of porous tritium-breeding lithium silicate pebbles, using the above-mentioned physics and mathematical models. The consequences of pebble fracture on the effective thermal conductivity of the pebble bed will be emphasized in the assessment, accounting for the multiple factors mentioned above. Through comparison with experimental data, understand the thermal transport mechanisms of the pebble bed, and clarify its thermal mechanical performance under multiple simultaneous influences. The results of the proposed work will strengthen the foundation of design for the magnetic confinement fusion solid breeding blanket system.
Thermal analyses of beryllide pebbles in water vapor atmosphere as advanced neutron multipliers
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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 reacted with water vapor at high temperature. In contrary to these Be pebbles, Beryllium intermetallic compounds (beryllides) are one of promising materials because of its much more stable chemical reactivity. Work 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. From the results of pebbles fabrication, as-received Be12Ti pebbles which consisted of Be, Be12Ti and Be17Ti2 phases, homogenized Be12Ti pebbles, and as-received Be17Ti2 pebbles have been successfully fabricated. Using these pebbles with a reference of Be pebbles, oxidation property, hydrogen generation, and reaction heat were investigated. Beryllides pebbles indicated much more resistant to oxidation by H2O, lower hydrogen generation and reaction heat than Be pebble. Among these beryllides pebbles, homogenized Be12Ti pebbles and as-received Be17Ti2 pebbles showed lower weight gain, hydrogen generation and reaction heat than as-received Be12Ti because as-received Be12Ti pebbles contains some fractions of Be phase inside pebbles.
P1.162

Parameter study on helium cooled ceramic breeder blanket neutronics with CFETR system code

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The development of system code for CFETR (China Fusion Engineering Test Reactor) is in progress for the optimization of the CFETR design in both core physics and engineering. As one of the key modules, the neutronics interface module has been implemented within the engineering framework of CFETR system code. The neutronics interface module, which is designed to work in conjunction with the general neutronics codes, is constituted by three sub-modules at present: the Tokamak fusion neutron source modeling sub-module, the neutronics modeling sub-module and the post processing sub-module. This work presents the parameter study on helium cooled ceramic breeder blanket neutronics with the neutronics interface module of the CFETR system code. The parameterized 3-D neutronic model of CFETR was set in the first place. Afterwards, the workflows of neutronic analyses were established. Several concerned geometric and material composition parameters of CFETR blanket were selected as arguments of certain ranges to pass the sensitivity analyses for target parameters (such as the tritium breeding ratio). And then, the workflows were abstracted as mathematical models and the so-called response surfaces for target parameters were established. Based on the workflow or the response surfaces, the fast estimation and optimization of target parameters could be achieved. Finally, an optimal set of parameters was proposed for CFETR helium cooled ceramic breeder blanket.
Chinese Fusion Engineering Test Reactor (CFETR) is an ITER-like fusion device that was proposed to achieve 200 MW fusion power, 30-50% duty time factor, and tritium self-sufficiency. As a candidate blanket concept for CFETR, a helium cooled solid breeder (HCSB) blanket was designed following the specific requirements. The helium cooling system (HCS) is an important ancillary system of HCSB blanket for CFETR. The preliminary design and description of HCS was already done while accident cases are investigated for the HCS. All components in the system are modeled as well as the main control strategy. Two typical operation modes for the HCS have been considered: a pulsed operation and a steady state operation. Three accident cases for each of the two operating modes are studied with RELAP5 including ex-vessel loss of coolant accident, loss of flow accident and failure of pressure control system. Ex-vessel loss of coolant accident caused by a double-ended pipe break of the HCS is considered as one of the most critical accidents. Loss of flow accident caused by the circulator seizure of the HCS is worth to study. The analysis of failure of pressure control system is critical for the pressure level in the system. Simulation results show that the design parameters of HCS is enough to sustain these accidents and to achieve the requirements of HCSB blanket. Key words: Safety analysis, RELAP5, HCS, HCSB blanket
P1.164

Thermo-Mechanical Analysis of the support component Between TBM and TBM-shield

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After the Conceptual Design Review (CDR), Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM) is in progress of the preliminary design phase. The detailed design work was performed on the connecting supports which are connected between the TBM and the TBM-shield. The geometric design of the connecting supports are referred from the connection design of the blanket first wall. The other types of design are H-beam and a flanged connection which are typically used in the industry. These connection types are based on the bolted joint. The thermo-mechanical analysis was performed characteristics of connecting support types according to the material and the loads conditions. The TBM is designed to make of a RAFM (a reduced activation ferritic-martensitics steel) while the material of TBM-shield is 316L(N)-IG. The material of the connecting support should be considered due the different thermal expansion. The concentrated stress would be generated on the dissimilar welded contact region. The pretention load on the bolted joint is major factor to determine the stress distribution on the connecting support at the specific loads condition. The design modification is continued to meet the design requirement in all loads conditions.
P1.165

**Accident Analysis on LOCA in HCCR-TBS towards CCWS-1**

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Korean Helium Cooled Ceramic Reflector (HCCR) Test Blanket System (TBS) will be operated at elevated temperature with high pressure helium coolant during normal operation in ITER. One of the main ancillary systems of HCCR-TBS is Helium Cooling System (HCS) which play an important role to extract heat from HCCR Test Blanket Module (TBM) by the helium coolant to keep the operational temperature and the extracted heat is finally transferred to ITER CCWS-1 (Component Cooling Water System) by a Printed Circuit Heat Exchanger (PCHE) in the HCS. In such circumstances if Loss Of Coolant Accident (LOCA) occurs in the PCHE, the high pressure helium coolant in the primary side goes into low pressure water in the secondary side thus pressurizing CCWS-1. In addition, since the helium coolant contains tritium due to permeation from the TBM, tritium migrates into CCWS-1, a non-nuclear system. In this paper, accident analysis for LOCA in the heat exchanger is presented. For the analysis, GAMMA-FR code which has been developed for fusion applications was used. Main components in the HCS and CCWS-1 were modelled as volume and junctions. The accident analysis was performed for the reference case with ten channels rupture and sensitivity study was also performed by changing the crack size. The results show that pressure and tritium requirement of CCWS-1 can be met in spite of LOCA in the heat exchanger of the HCCR-TBS HCS.
P1.166

**Development of the real-scale helium circulator for the HCS of HCCR-TBS**

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A helium circulator, to provide up to 1.5 kg/s of helium flow with pressure of 8 MPa, has been developed for the HCCR-TBS. To overcome the pressure drop of the helium cooling system of the HCCR TBS, the circulator is designed maximum speed of 70,000 RPM with electric power of 150 kWe to meet compression ratio of 1.1. One of the major design features of the circulator is that the impeller and the shaft are mechanically separated to isolate helium coolant pressure boundary from the atmosphere. The rotational momentum of the shaft, however, is transfer to the impeller by the magnetic coupling device. The circulator will be installed in the HeSS facility by May 2016 and the performance test will be performed to verify design parameters and performance of the circulator.
Thermal optimization of the Helium-Cooled Lithium Lead breeding zone layout design regarding TBR enhancement

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Within the framework of EUROfusion R&D activities CEA-Saclay has carried out an investigation of the thermal and mechanical performances of alternative designs intended to enhance the Tritium Breeding Ratio (TBR) of the Helium-Cooled Lithium Lead (HCLL) blanket for DEMO. Neutronic calculations performed on the 2014 DEMO HCLL layout have indeed predicted a value of TBR equal to 1.07, lower than the required value of 1.1, necessary to ensure the tritium self-sufficiency of the breeding blanket taking into account uncertainties. In order to reach the TBR target, the strategy of the steel amount reduction inside the HCLL module breeding zone has been followed by suppressing some stiffening/cooling plates inside the BZ. Since all the plates inside the BZ are actively cooled by helium, each change in their geometric layout has a strong impact on the thermal response of the module. Moreover, the removal of stiffening plate may impact the resistance of the box in case of in-module’s loss of coolant. In order to optimize from the thermal point of view the HCLL BZ layout, attention has been paid to the outboard equatorial module of the breeding blanket and the thermal behaviour of different geometric layouts of the elementary Breeding Unit has been assessed with the aim of checking that the thermal requirements foreseen for the EUROFER steel structural material are met while respecting acceptable pressure drops. Mechanical calculations have also been performed to analyse the behaviour of the module in faulted condition without full vertical stiffening plates. To perform this research campaign a theoretical-numerical approach, based on the Finite Element Method (FEM), has been followed and the qualified Cast3m FEM code has been adopted. Results obtained have been herewith presented and critically discussed, highlighting the open issues and suggesting the pertinent modifications to DEMO HCLL module design.
Influence of modifications of HCLL blanket design on MHD pressure losses

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In 2008-2009 experiments have been performed to investigate liquid metal magnetohydrodynamic (MHD) flows in a scaled mock-up of a helium cooled lead lithium (HCLL) blanket. In order to improve the mechanical stiffness of the blanket module the design of the stiffening plate between two hydraulically connected breeder units (BUs) has been later modified. In the former design the liquid metal passed from one BU to the adjacent one by flowing through a slot that extended along the entire width of the BU. In the most recent design this opening has been replaced by a series of smaller gaps. Therefore the velocity increases locally owing to the reduced cross-section along the flow path and the liquid metal has to expand along magnetic field lines to enter the next BU. These flow conditions are known to create additional pressure losses as a result of the occurrence of 3D MHD phenomena and significant inertia effects. In order to quantify the influence of the design modifications described above, the available test section has been adapted to the new design features. Experiments have been performed to record pressure distribution in the new mock-up in a wide range of flow parameters and data have been compared with results obtained by using the former test section. Experimental results show that these design modifications near the first wall lead to a local increase of pressure drop by a factor 3-3.5 compared to previous data. As a consequence the total pressure drop becomes larger too. Additional pressure losses near the first wall, where the fluid expands and contracts along magnetic field lines, seem to be mainly related to inertia effects confined in boundary layers along walls parallel to the magnetic field.
The current status and challenges of the PbLi cold trap development undertaken at CVR

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Research Centre Rez (CVR) is actively involved in research and development of a purification technique of the liquid lithium-lead eutectic alloy based on use of a cold trap. The first activities linked to this field are dated since 2003. They are carried out within the major European fusion projects (F4E, EFDA and EUROfusion) and the Czech national CANUT project. For the cold trap development, the MELILOO experimental PbLi loop is used. The presentation will describe evolution of the cold trap design covering all the concepts studied and experimentally tested in the past as well as the ones which are currently being developed in CVR and will be commissioned in the near future. Covered are also technical issues which have been met and solved during the process of the cold trap development. A significant part of the article deals with description of the latest air cooled cold trap concept. Detailed description of the design, operational experience and basic results of the engineering analyses with focus on the thermal-hydraulic studies are included. A series of simulations to evaluate velocity and temperature field of the PbLi liquid metal inside the cold trap was performed for different operational conditions using CFD approach. The results of these studies were subsequently compared with the experimental data and the suitability of the used computational code was evaluated. Based on the results of the engineering analyses and operational experience, steps leading to improvement of efficiency and reliability of the device were proposed and are discussed in the paper.
P1.170

Helium bubble release from Pb-16Li within the Breeding Blanket

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In a prospect of future fusion power plants construction, different concepts of tritium breeding blankets are being developed within the EUROfusion breeding blanket work package. Three main concepts using Pb-17Li as breeder, the HCLL (Helium Cooled Lithium Lead), WCLL (Water Cooled Lithium Lead) and DCLL (Dual Coolant Lithium Lead) are developed as candidate technologies for European DEMO facility. Helium is a by product of tritium breeding by lithium decomposition. Compared to tritium, its solubility is several orders of magnitude lower. Previous experiments have shown, that bubble formation within the PbLi alloy during irradiation may occur and suggested they pose a serious threat for breeding blanket concepts based on liquid PbLi. An important aspect of the helium bubble formation is tritium desorption into the gas phase formed within the liquid metal. This part of the tritium will be removed along with the helium and will not enter the Tritium Extraction System. To assess the extent of the helium gas release, a simplified model of a single sector of the breeding blanket had been developed and local helium production rates have been calculated. Based on current physical properties and recent development of the blanket geometry, it is possible to estimate the local gas bubble release rate within each module, the gas bubble release profiles along the PbLi conduits and the gas composition.
Nuclear analysis of the HCLL blanket for the European DEMO

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The EUROfusion Consortium aims at developing a conceptual design of a fusion power demonstrator (DEMO). The breeding blanket facing the plasma is one of the key components of DEMO. It must ensure tritium self-sufficiency and heat removal functions. The Helium Cooled Lithium Lead (HCLL) blanket concept is one of the four breeding blanket concepts investigated for DEMO. 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. Within the EUROfusion organisation CEA, with the support of Wigner-CR and IPP-CR, is in charge for the design of the HCLL blanket. This paper presents the nuclear analysis carried out for a new DEMO baseline with HCLL blanket. This baseline has a bigger minor radius and a smaller divertor that give extra space for the breeding blanket. The tritium breeding ratio obtained with the previous baseline and the HCLL blanket was below the 1.1 target (1.08). Different HCLL breeding blanket design options were investigated to increase the TBR. The helium manifold architecture was improved to increase the breeding zone. The number of cooling plates was reduced. The vertical stiffening plate removal was studied. The objective was to reduce the steel amount in the breeding zone. The TBR achieved with these new HCLL designs and new DEMO baseline present comfortable margins (1.15 to 1.23). The studies of DEMO-HCLL-2014 showed a nuclear heating in coils (100 W/m³) above the limit (50 W/m³). To improve the radiation stopping power of the HCLL blanket a shielding block (in Eurofer and other investigated material) is added in the manifold. The shielding block volume is taken from the breeding zone and the manifold region. Finally, the obtained HCLL design is a compromise between the tritium production and the shielding capability.
The dual functional lead lithium (DFLL) test blanket module (TBM) concept has been proposed by FDS team to demonstrate the techniques basis of DEMO liquid blanket concepts, including quasi-statistic lead lithium (SLL) breeder blanket and the dual-cooling lead lithium (DLL) blanket. In recent years, series R&D work for DFLL-TBM carried out are mainly on five topics: 1) Structural materials (i.e. CLAM steel) qualification; 2) PbLi/He coolant technologies and safety issues; 3) RAMI (Reliability, Availability, Maintainability and Inspectability) estimation of DFLL blanket; 4) Small TBM mockup neutronics experiments; 5) Tritium behavior in CLAM steel. The latest progress is as follows:

The fabrication technologies were mature and the properties basically met with the requirement of TBM. The double-coolant multi-function experimental loop DRAGON-V is being built and the construction will be finished at the end of 2016 and then the out of pile tests can be performed with it. On the other hand, the construction of the high intensity neutron generator (HINEG) for D-T fusion neutrons has been finished and the experiments related to the D-T fusion neutrons and materials are being carried. The RAMI analysis was performed on the conceptual design of DFLL-TBM. The inherent availability of the DFLL-TBM system after implementation of mitigation actions for total 181 failure modes was calculated to be 98.57% for 2 years’ operation. A small blanket mock-up has been fabricated and the preliminarily D-T neutrons irradiation results validated the precision of the data library and the reasonableness of the TBM design. Permeation experiments of hydrogen and deuterium has been performed, the transport parameters and results were obtained and consistent well with the results of other RAFM steels.

Based on the latest R&D progress, the status and strategy of DFLL blanket for DEMO are presented in the paper. Keywords: Fusion Reactor; Liquid Breeder Blanket; DFLL-TBM; Development Roadmap
Repeatability of irradiation damage and of recovery by post-irradiation annealing of EUROFER base steels

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Former Investigations clearly had revealed that embrittlement and hardening of RAFM steel after 15 - 70 dpa neutron irradiation damage remarkably can be reduced by short time post-irradiation annealing (PIA) at 550 °C [1, 2]. The purpose of this study is to demonstrate the repeatability of the damage- and recovery-mechanisms to RAFM 7-10% CrWVTa, ODS EUROFER, Boron doped heats of the prior 330 °C Bor 60 reactor (SSC RIAR) irradiations CP1 (6.5 dpa), CP2 (2.5 dpa), and WTZ RUS 01/577 (15 dpa). Tensile and KLST Charpy-V impact specimens have been investigated in irradiated state after a first annealing (PIA-1), after a subsequent further irradiation up to 4.6 – 5.4 dpa at 330 °C in the Bor 60 reactor, and after a second annealing (PIA-2). Both, tensile and impact tests clearly indicate that irradiation induced hardening occurs after the second irradiation of intermittently annealed (PIA-1 and thus recovered) material in the same order like after the first irradiation. In addition, a subsequent PIA-2 leads to a complete recovery of the newly applied neutron damage. This suggests that lifetime of fusion power plant components of RAFM steels substantially could be extended by repeated intermediate annealing treatments and respectively that components could withstand much higher nominal damage dose rates using intermediate annealing. [1] Gaganidze, E.; Petersen, C.; Materna-Morris, E.; Dethloff, C.; Weiß, O.J.; Aktaa, J.; Povstynako, A.; Fedoseev, A.; Makarov, O.; Prokhorov, V.: Mechanical properties and TEM examination of RAFM steels irradiated up to 70 dpa in BOR-60, J. Nucl. Mat. 417(2011), p 93 - 98 [2] Sacksteder, I.; Schneider, H.-C.; Materna-Morris, E.: Determining irradiation damage and recovery by instrumented indentation in RAFM steel, J. Nucl. Mat. 417(2011), p 127 - 130
ODS ferritic steels obtained by STARS, an innovative processing route without mechanical alloying

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Oxide dispersion strengthened ferritic steels (ODS FS) are candidate structural materials for future fusion reactors thanks to their high temperature strength, high creep resistance, and good resistance to neutron radiation. Their outstanding behavior is a direct consequence of their extremely fine microstructure and the presence of highly stable and finely distributed nanometric oxide precipitates. The conventional processing route of ODS FS includes mechanical alloying (MA) of elemental or gas atomized prealloyed powders with Y₂O₃ particles followed by consolidation by hot isostatic pressing (HIP) or hot extrusion, and finishes with thermo-mechanical treatments to obtain fine grain structures with very fine Y-Ti-O nanoclusters dispersion. However, MA involves several drawbacks like contamination from grinding media, and batch to batch heterogeneities. To avoid MA, a new route called STARS (Surface Treatments of gas Atomized powder followed by Reactive Synthesis) has been developed and is the core of the present work. This route is inspired in the GARS method (Gas Atomization Reactive Synthesis) developed by I.E. Anderson in AMES laboratory. FS powders already containing the oxide-dispersion formers (Fe-14Cr-2W-(0.3-0.56)Ti-(0.18-0.37)Y) were obtained by gas atomization. Then, a metastable oxide layer was formed on the surface of powder particles. When HIPped at elevated temperatures (>1220°C), this oxide layer dissociates and Y-Ti-O nano-oxides precipitate in the ferritic matrix, as observed by TEM and XAS (X-ray Absorption Spectroscopy). Post-HIP heat treatments at elevated temperatures dissolved the remaining oxides located at prior particle boundaries. However, they can be detrimental from the microstructural point of view as they favour thermally induced porosity and coalescence of residual carbonitrides. Subsequently, hot rolling was performed at 1050 °C. Finally, heat treatments at different temperatures were performed on hot rolled material to remove residual stresses and promote recrystallization. This work concludes that the STARS route has great potential to obtain ODS Ferritic Steels.
Assessment of F82H development status toward DEMO with respect to the existing design code

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F82H is the reduced activation ferrite/martensitic (RAFM) steel which has been developed in Japan. Its chemical composition was designed based on the composition of high Cr heat resistant steel, Mod9Cr-1Mo, reducing activity level by replacing Mo to W, Nb to Ta, and reduce N level to suppress $^{14}$C formation. In order to prove its potential as the structural materials, it is critical to provide data of various properties in accordance with the requirements defined in the existing design code. Database preparation status was assessed by referring to the dataset provided to define Mod9Cr-1Mo in ASME and RCC-MRx. It was pointed out that some physical properties were not sufficiently evaluated, and some gaps in the dataset of the major mechanical properties were identified even though the majority of it fulfils the requirements. On the other hand, the gap analyses indicated that there are numerous undefined properties and fabrication technologies in the existing design code in view of fusion DEMO application. The magnetic properties are the essential properties to evaluate the impact of plasma disruption on the blanket structure, but no definition was found in the existing design code. HIP joining is selected as the candidate method to fabricate the breeding blanket first wall with cooling channels, but no standard is available for HIP joining method to use it for highly pressurized structure. Irradiation effects have been recognised as the potential risks for securing the structural integrity for the lifetime of the fusion blanket system. Thus, the irradiation database has been accumulated for various properties, but the number and quality of the data is limited because of the lack of fusion neutron source, irradiation volume and the standard for small specimen testing method. The design rules also need to be developed to compromise those degradation of properties due to irradiation.
Effect of irradiation hardening on deformation behavior of blanket structures fabricated by F82H

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The box structure of water-cooled solid breeding (WCSB) blanket fabricated by F82H is being developed in Japan for the DEMO reactor. In the DEMO operation, the structural materials in the region of first wall (FW) will be exposed to severe fusion neutron irradiation. One of the issues is the loss of ductility for the structural materials due to severe fusion neutron irradiation. In the case of in-box loss of coolant accident (LOCA), the pressure of the pressurized water reactor (PWR) will be loaded inside the box structure and then cause the large deformation of the structural materials. The objective of this work is to estimate the effect of irradiation hardening on deformation behavior of the blanket structures under the internal pressure assuming in-box LOCA. Structural analysis of the box-shaped blanket with a surface crack at the coolant corner was conducted by elastic-plastic finite element analysis (FEA). True stress vs. true strain curve of F82H-IEA-heat tested at 300 °C was employed for the FEA. The curve for irradiated F82H-IEA-heat was estimated based on engineering stress-strain curve and true fracture strain estimated from reduction of area. The Isotropic hardening rule was employed for the plastic deformation behavior of the materials. The internal pressure of 15.5 MPa was applied to cooling channels and the inner surface of the box-shaped blanket. FEA was conducted in order to evaluate the crack-tip stress fields. Irradiation hardening of structural materials suppressed crack-tip opening under applied internal pressure. However, the high stress triaxiality factor (STF) for irradiated materials was widely distributed ahead of crack-tip compared to that for unirradiated materials. Therefore, irradiation hardening was prone to accelerate the ductile crack propagation. Further analysis on the crack initiation and propagation behavior will be discussed in this paper.
Characterization of F82H HIP joints by acoustic emission

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The hot isostatic pressing (HIP) is the key technology to fabricate the first wall of the fusion blanket system. Generally, the Charpy impact test is applied to evaluate the failure behavior of the HIP joint however there is a drawback that this cannot be applied to the practical thin-walled first wall component since the Charpy impact test requires a long bar specimen. Alternatively the authors recently proposed the torsion test method to assess the HIP joint. The torsion method has an advantage that the test specimen can directly be obtained from the component. Besides a good correlation with the test results by the Charpy impact test was shown. The torsion method is therefore considered as a promising standard technique in qualification of the HIP joint. The performance of the HIP joint depends on the presence of sub-micron precipitates distributed on the HIP joint. However, the detailed failure mechanism of the HIP joint with or without these precipitates is still uncertain and these are primarily evaluated in this paper. To detect any failure signals during the test, the acoustic emission (AE), which has widely been used in determination of the failure location, failure mode, failure energy, etc., was first applied in the developmental torsion test. Most of AE signals were detected upon visible cracking beyond the maximum torque applied. A slight difference in the accumulation process of the AE signals was found between the bulk and the HIP joint material. In failure of the bulk material, continuous and relatively high energy AE signals were identified, while low energy AE signals were typical for the HIP joint material. Such difference is possibly due to the varied failure modes closely related to the presence of precipitates. The detailed failure behavior focusing on the AE energy will be discussed for various HIP joint materials.
P1.178

**Ion irradiation effects on microstructure and mechanical properties of VPS-W-coated F82H modified by FSP**

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Reduced activation ferritic/martensitic steel, as typified by F82H, is a promising candidate for structural material of DEMO fusion reactors. To prevent plasma sputtering, tungsten (W) coating was essentially required. Vacuum plasma spray (VPS) is one of candidate coating processes, but the key issues are the degraded mechanical and thermal properties due to its relatively higher porosity and smaller density. Friction stir processing (FSP) was applied on VPS-W to solve the issues, and successively improved its hardness and thermal conductivity in unirradiated condition. Fine-grain microstructures induced by FSP would be a primary reason of this improvement. These structures were observed not only in VPS-W but also in F82H substrate, and it is expected that both VPS-W coating and substrate F82H could improve their irradiation-tolerance. This study aims to examine the irradiation effects on hardness and microstructure of VPS-W coated F82H modified by FSP, with a special emphasis on tolerance of void swelling in F82H substrate and W-F82H interface. F82H IEA-heat was used as a substrate in this study, 0.5 or 2 mm-thick W was coated on F82H by VPS and then modified with FSP. 6.4 MeV Fe\(^{3+}\) and 1.0 MeV He\(^+\) irradiation with 50 dpa at 470°C at the DuET facility in Kyoto University. Nano-indentation tests were performed to evaluate hardness after the irradiation. Microstructure was characterized by OM, FE-SEM, and FE-TEM. After dual ion irradiation to 20 dpa at 470°C, the void swelling of 0.48% was measured in F82H-IEA with the diameter of 5.19 nm and the number density of \(4.18 \times 10^{22}/m^2\). Contrarily, no void was detected in FSPed-F82H at the same condition. From a microstructural point of view, it was suggested that lots of sinks introduced by plastic flow during FSP could decrease void swelling under the condition.
Deformation of dissimilar-metals joint between F82H and 316L in impact tests after neutron irradiation

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Connection between blanket and out-vessel component is essential to fusion reactors. In the present study, electron beam welding was carried out to fabricate a dissimilar-metals joint between a blanket structural material, F82H steel, and an out-vessel component material, 316L steel. Impact properties and deformation behavior of the joint were analyzed after neutron irradiation. Two types of Charpy V-notch (CVN) impact specimens were used, i.e. 1/3 CVN and 1.5 CVN specimens. The size of the former is 25.4mm×3.3mm×3.3mm; the latter is 20mm×1.5mm×1.5mm. The V-notches were placed at the weld metal (WM), the heat affected zone (HAZ) of F82H, and the base metals (BMs). SSJ type tensile specimens with a gauge size of 5mm×1.2mm×0.35mm were also prepared. Neutron irradiation was carried out on the 1.5 CVN and the tensile specimens with Belgian Reactor II at 300°C up to 5.6±0.1×10²³ n m⁻² (E > 1MeV). Impact tests, tensile tests, and hardness tests were conducted after the irradiation. Absorbed energy for the joint notched at WM and HAZ of F82H was 77 and 90 J cm⁻², respectively, in the impact tests for the 1.5 CVN specimens at room temperature. While, absorbed energy after the neutron irradiation was 74 and 67 J cm⁻². Tensile tests and hardness tests exhibited neutron irradiation hardening at the whole part of the joint. The hardening ranged from 22 to 200 VHN and was maximized at the HAZ of F82H. Even the large hardening at the HAZ did not lead to significant degradation of absorbed energy. Microstructural observation indicated that the deformation of the joint is asymmetric and localized in 316L steel. The deformation in 316L likely induced stress relaxation around the WM and HAZ, and was effective to maintain absorbed energy. The asymmetric deformation will be analyzed by using finite element method simulation combined with mechanical property tests.
Heavy ion irradiation technique has been used for simulating fusion neutron irradiation on materials. However, mechanical testing technologies were limited due to the thin irradiated layer only up to several um in depth. Nanoindentation hardness were often used for evaluating irradiation hardening behaviour of ion-irradiated subsurface. This study investigates micro-pillar compression behavior of ion-irradiated F82H reduced activation ferritic steel. Fe$^{3+}$ ion beam irradiations were carried out by using TIARA facility at 300 degree C up to 3 dpa. Micro-pillar specimens of 1, 3, and 10 um in diameter on the unirradiated surface and 1 or 3 um for the ion-irradiated surface were fabricated by focused ion beam (FIB). Compression tests on the micro-pillars were performed by nanoindentation device equipped with a flat punch diamond tip. Yield stress of unirradiated specimens increased with decreasing specimen size. By using the 1 um micro-pillar, irradiation hardening were successfully evaluated as increase of yield stress. Deformation behavior of micro-pillars before and after the ion-irradiation showed multiple gliding at a same slip system like a single crystal.
P1.181

Evaluation of Tensile Properties of F82H Welded Joint Using Small Punch Test

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The small punch (SP) test method is one of the small specimen test techniques (SSTT). This method has several advantages: it requires only a small specimen, its test method is simple, and it is able to evaluate various mechanical properties. For these reasons, the SP method is commonly used in post-irradiation testing (PIE) of nuclear materials and as a damage evaluation technique for actual structural materials. However, because of the intricate deformation behavior of the specimen, stress and strain cannot be easily calculated; thus, it is difficult to make a direct comparison with standard test results. In this study, we attempted to estimate tensile properties by simulating SP test results, using Finite Element Analysis (FEA). By adjusting the material property parameters defined by the Ramberg-Osgood law in the FEA, the load-displacement curve of the SP test results were matched with that of the FEA simulation results. The actual stress-strain curves were reproduced using the obtained Ramberg-Osgood law, and it used to estimate the tensile properties up to the homogenous deformation region. This method was applied to an SP specimen collected from the local region of reduced activation ferrite/martenstic steel F82H welding material, and it was used to evaluate the tensile properties of the welding base material, the weld metal, and the heat-affected zone.
The R&D of high performance fuel cladding materials has been considered to be essential for the realization of fusion and Gen IV fission energy systems. The 9Cr oxide dispersion strengthened (ODS) martensitic steels was developed for applying as cladding materials of sodium-cooled fast breeder reactors (FBRs). The steels exhibited good compatibility with sodium, while the corrosion resistance was not good enough in supercritical pressurized water (SCPW) and lead-bismuth eutectics (LBE). High-Cr ODS ferritic steels added with Al showed a drastic improvement in the corrosion resistance in SCPW and LBE. ODS steels are usually produced by powder metallurgy techniques involving a series of steps: (i) mechanical alloying (MA) of a powder mixture, (ii) canning and degassing of the powder, (iii) consolidation of the powder particles by hot extruding, and (iv) normalizing and tempering heat treatment of the consolidated products. The material performance of ODS steels is controlled by the dispersion morphology of dispersoids that is significantly influenced by the processing conditions not only during heating but also during milling. Four different ODS ferritic steels, Fe-16Cr-(0/4)Al-0.1Ti-0.35Y2O3, in wt%, were prepared by MA of elemental powders with Y2O3 particles either in attritor mill or in planetary ball mill, and consolidated by hot extruding. Milling in air is effective to reduce the particle size of MA powder. SEM and PSD analyses showed that the processing capacity of powder particles during planetary ball milling has improved twofold compared to our previous condition. In this study, recent experimental results on the effects of MA parameters, such as milling atmosphere, device, and Al addition, on the microstructure and Charpy impact properties of the ODS ferritic steels will be shown to conclude that lowering the excess oxygen and nitrogen contents in steels is the critical issue for enhancing their impact properties.
Tensile properties of F82H steel after aging at 400 to 650°C for 30,000 h

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A reduced-activation ferritic steel, F82H steel, is the primary candidate structural material for fusion blanket. It has been clarified that long term aging degrades both strength and ductility due to precipitation of Laves phase (Fe2W) and other changes in microstructure. In order to evaluate the degradation and to clarify its mechanisms, the present study analyzed the tensile properties of F82H after long term aging at the operation temperature. The material used was F82H-IEA heat with a composition of Fe-7.71Cr-1.95W-0.091C-0.16V-0.02Ta-0.11Si-0.16Mn-0.002P-0.002S-0.006N. The final heat treatment conditions were normalizing at 1040°C for 40 min and then tempering at 750°C for 1 h. F82H was aged from 400 to 650°C for 1,000 to 30,000 h. Type SS3 tensile specimens with a gauge size of 5 x 1.2 x 0.75 mm were machined before and after the aging. Tensile tests at room temperature (RT) were conducted in the air, while high temperature tests were performed from 400 to 650°C in a vacuum. Ultimate tensile strength (UTS) at RT before aging was 673 MPa. UTS after aging at 500, 550, 600 and 650°C for 1,000 h was 653, 652, 651 and 583 MPa, respectively. Since the scattering of the strength is about 50 MPa in tensile tests, only the change, -90 MPa, in UTS at 650°C is recognized as obvious degradation. Degradation more than -50 MPa was observed at 650°C for 30,000 h aging, while it was detected at 600 and 650°C for 10,000 and 30,000 h aging. In conclusion, the degradation ranged from -57 to -176 MPa, and increased with increasing aging temperature and time. The mechanisms for the change in tensile properties will be discussed.
P1.184

Ultrasonic testing of different shapes of support rib and lower bracket assembly weld joints

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ITER Vacuum Vessel (VV) is made of double walls connected by ribs structure and flexible housings, space between these walls is filled up with In Wall Shielding (IWS) blocks to (1) reduce neutrons streaming out of plasma and (2) reduce toroidal magnetic field ripple. These blocks will be connected to the VV through a supporting structure of Support Rib (SR) and Lower Bracket (LB) assembly. SR and LB are two independent components manufactured from SS316L (N)-IG material. Two horizontal lower brackets will be welded with one vertical support rib to make an assembly. These are welded by Tungsten Inert Gas (TIG) and Shielded Metal Arc Welding (SMAW) as full penetration joint and tested using various NDT methods such as Liquid Penetrant testing (LPT) and Ultrasonic Testing (UT), etc. There were many challenges while qualifying UT technique such as, 1) Interpretation due to Coarse Grain Structure of the material and weld 2) The limitation of scanning distance which required special probes for testing. There are two shapes of plates, (i) rectangular (ii) trapezium welded with rectangular plate with straight weld line. All essential requirement related to qualification of personnel, calibration of instruments, probes and design of mock-up calibration blocks were identified prior to establishment of the UT procedure. Two calibration blocks are manufactured based on the geometry of the weld joints. Artificial defects (2mm and 3mm diameter side drilled holes and 0.7mm width notch) were inserted into the blocks so as to correlate the same with the actual defects. The validation of procedures was carried on calibration blocks using Radiography testing and Macro Analysis Test. Approximately 170 Nos of such welded assembly have been tested successfully. This paper detailed the work that has been carried out to establish ultrasonic technique for weld joints of Support Rib and Lower Bracket Assembly.
P1.185

Long-lived radionuclide activity formed in ITER steel composites in 6Li-D converter neutron field

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Measurement and calculations of long-lived radionuclide activity forming in the 14 MeV neutron field, in 6Li-D converter were done, in some steel composites of ITER. The activation was conducted in September, 2014 in the thermal-to-14MeV neutron converter constructed in National Centre for Nuclear Research in Poland. This irradiation facility was placed in the core of MARIA research fission reactor. The activation lasted 135 hours. The steel samples were of 10x10x1 mm plate-shape and of approximately 0.8 g mass. The neutron spectrum in 6LiD converter was assessed as a combination of MCNP simulated spectra and tailored to the measured activity of activation foils. The very high threshold (ca 15 MeV) reactions, Au-197(n,3n)Au-195 and Bi-209(n,3n)Bi-207 were especially useful as indicator of converter neutrons. Fast neutron fraction (above 1 keV) was 53% and fraction of fast neutrons from 6LiD converter in the energy range above 12.5, 13.5 and 14.2 MeV was respectively 80, 97 and 98.7%. Neutron density in the same energy ranges was respectively 3.5, 3.1 and 2.1 10⁹ cm⁻²·s⁻¹. The activity measurement of the radionuclides formed in steel composites was done by means of gamma-ray spectrometry. Activity calculations were done by means of FISPACT-2010 using the activation library EAF-2010 and assessed neutron flux. Total activity of measured gamma emitting radionuclides in steel samples after 20 days of cooling was in the range 185 – 208 MBq/g. The calculated to experimental values ratio (C/E) differs for particular radionuclides and are in the range 0.63-0.86 for Cr-51, 0.72-0.83 for Mn-54, 0.68-0.85 for Co-58, 1.08-1.24 Fe-59, 1.21-1.40 for Co-57 and 0.0-2.54 for Co-60. Carried out measurements and calculations confirm the presence of high fusion neutron flux and proves that tested converter is one of the strongest available continuous 14 MeV neutron source.
Study of Magnetic Properties of IWS Materials during Material and Component Manufacturing

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In wall Shielding blocks will be inserted between inner and outer shell on ITER Vacuum Vessel (VV) and will fill up about 60\% of volume between two shells. IWS blocks comprise of number of plates stacked together with fasteners. There are two types of IWS blocks, (i) Primary IWS blocks made of Austenitic stainless steels (SS304B4 and B7) to provide neutron shielding to all components inside cryostat, particularly to Toroidal Field (TF) coils and (ii) Ferromagnetic inserts (FM) made of Ferritic steel (SS430) to reduce TF ripple at mid-plane of plasma. Critical material requirement for FM blocks to achieve 0.3\% TF ripple at mid-plane is saturated magnetization which should be 1.6-0.166 T at room temperature at 80,000 A/m field. SS430 is selected for FM blocks and its chemical composition is adjusted to achieve this value. Chromium was varied from 16\% to 17\% and this paper describes the variation in magnetic properties of SS430 with this variation in chemical composition. Another important requirement for all non-FM blocks is that the magnetic permeability of, (i) all raw materials and (ii) of all components after machining must not exceed 1.03. During manufacturing IWS Components undergo many operations like Water Jet cutting, CNC machining, Turning, Rolling, Forging, Solution Annealing etc. which may change the magnetic properties of materials from original value of 1.03. With appropriate selection of materials of various tools and fixtures and adjusting different parameters (e.g. cutting speed, thread rolling speed, welding etc) magnetic permeability was retained to almost to its original value. This paper describes the variation in magnetic properties of IWS materials during material production as well as the machining of IWS components from these materials.
Qualification of structural stainless steel products for the ITER correction coil cases

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The ITER Correction Coils (CCs) consist of three sets of six coils, Bottom (BCC), Side (SCC) and Top Correction Coils (TCC), respectively, located in between the toroidal (TF) and poloidal field (PF) magnets. The CCs rely on 10 kA NbTi Cable-in-Conduit Conductor (CICC). Each CC winding pack is enclosed inside a 20 mm thick stainless steel case, providing structural reinforcement against the electromagnetic loads arising in the winding pack. BCC and TCC cases are designed with a U-shaped cross-section and a cover, while SCC case consists of two L-shape half cases. The material selected for the coil cases is the austenitic stainless steel 316LN. It shall feature ready weldability both by laser and conventional techniques, high strength and toughness at 4 K. Material production involves not only hot rolled plates of different thicknesses, but also heavy gauge extruded L and rectangular hollow shapes. An adapted steelmaking route including Electroslag Remelting (ESR) combined with a hot transformation step involving redundant multidirectional forging of the slabs and of the billets was necessary to fulfil the stringent material specification and confer cleanliness, fineness of the structure and homogeneity to the final products. An extensive follow-up of the steel manufacturing through systematic non-destructive and destructive examinations was carried out in order to ascertain the soundness and the homogeneity of the final products. Starting from the results of the quality controls performed, the paper highlights the progresses accomplished throughout the steel manufacturing and discusses the properties achieved on the final products that were enabled by the selected manufacturing route. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.
A comparison of properties of ARAA produced by VAR and ESR refining methods

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Reduced activation ferritic-martensitic (RAFM) steel is considered a primary candidate for the structural material in a fusion reactor. The operational design window for a blanket is limited by the high-temperature creep and low-temperature irradiation embrittlement of the structural material, and it is therefore essential to develop RAFM steel which can withstand high temperatures and high energy neutron irradiation. For this purpose, an advanced reduced-activation alloy (ARAA) containing 0.01 wt.% Zr has been developed for structural material in fusion reactors in Korea. A five-ton scale heat of ARAA was successfully produced by the vacuum induction melting (VIM) and electro-slag re-melting (ESR) methods, for which thermal, physical, magnetic and mechanical properties were evaluated. Recently, a six-ton scale heat was produced by VIM but refined by different method, vacuum arc re-melting (VAR), and its properties were evaluated. Comparison of the properties of both ARAA heats with different refining methods reveals that strength and ductility of ARAA heat refined by VAR methods are much better than that by ESR, while thermal, physical and magnetic properties of the heat are comparable to those refined by ESR. The VAR heat contains less impurity than those refined by ESR, and exhibits more homogeneous microstructure. It is therefore concluded that VAR method is a better refining method for production of ARAA.
Chemical compatibility between ARAA alloy in lithium meta-titanate breeder material

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Chemical compatibility between Korean reduced activation ferritic-martensitic alloy (ARAA) and lithium meta-titanate breeder was investigated under operation conditions; high temperature and helium purge gas including low concentration of hydrogen. ARAA specimens were embedded inside lithium meta-titanate powder and compacted under the load of 200 MPa to form block-shaped samples. The samples were heated at 550 °C for up to 1000 h under helium with up to 1% hydrogen atmosphere to simulate breeding blanket environment. The surface of ARAA was chemically reacted with lithium meta-titanate breeder to form a chromium/iron oxide layer. The thickness of the reaction layer increased as the dwelling time increased. In this paper, the effect of hydrogen in the helium gas on the characteristics of the oxide layer was investigated. Microstructure, elemental distribution, and phase evolution of the reaction layer were analyzed using X-ray Diffraction (XRD), Scanning Electron Microscopy (SEM), and Electron Probe Microanalysis (EPMA). The growth mechanism of the oxide layer was evaluated.
Reheating cracking susceptibility in the weld heat-affected zone of reduced activation ferritic-martensitic steels

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Reheating cracking susceptibility in the weld heat-affected zone (HAZ) of reduced activation ferritic-martensitic (RAFM) steels was explored by evaluating stress-rupture parameters (SRP), which depends on rupture strength and ductility. The HAZs simulation and stress-rupture experiments were carried out using a Gleeble simulator at various temperatures, corresponding to post-weld heat treatment (PWHT). After stress-rupture tests, fracture morphologies and cross-sectional microstructures were observed by a scanning electron microscopy (SEM) and a transmission electron microscopy (TEM). The results revealed that the reheating cracking occurred in the vicinity of the intergranular Cr23C6 precipitation, due to the formation of Cr-denuded zone along the grain boundary.
Effect of Ti on microstructures and mechanical properties of reduced activation ferritic-martensitic (RAFM) steels

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The effect of addition of Ti on microstructures and mechanical properties in RAFM steels were investigated. Ti-bearing RAFM steels, designed based on the thermodynamic calculation, were fabricated by vacuum induction melting and hot-rolling process. All samples were heat treated by normalizing and tempering, resulting in tempered martensite with M23C6 carbides and MX precipitates. The microstructures were analyzed using optical, scanning electron and transmission electron microscopies quantitatively as well as qualitatively. Mechanical properties of the samples were evaluated by Vickers hardness, Charpy impact and tensile tests at various temperatures. The results indicated that Ti-added RAFM steels have enhanced tensile properties without sacrificing impact toughness due to precipitation of nanometer-sized (Ti,W)C carbides and decrease in the average size M23C6 of particles compared to the conventional RAFM steels.
Sample holder design for effective thermal conductivity measurement of pebble-bed using laser flash method

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The property of functional material for the design of the breeding blanket is very essential. Since the stress due to the thermal load on breeding blanket structure is one of the main design driver, the thermal property of the material is very important for thermal-structural and thermo-hydraulic analysis. In particular, the thermal conductivity is one of necessary input data for these analyses performed in order to understand the heat transfer phenomena and estimate the thermal stress. Since for the functional materials of solid type breeding blanket a pebble-bed form is mainly adopted instead of a bulk form such as a block or a disk, it should be needed to measure the thermal conductivity of pebble-bed. In this study, the effective thermal conductivity of pebble-bed is measured by laser flash method, which is one of the various thermal conductivity measurement methods, because this method has several advantages such as a wide thermal conductivity range of the measurement and a small amount of pebbles. A sample holder considering the heat transfer mechanism from the laser source to pebble-bed has been specially designed in order to apply the laser flash technique to the pebble-bed sample and it has been validated by the experiments. This paper introduces preliminary results of the effective thermal conductivity on the pebble-bed using this sample holder.
P1.194

Status and strategy of the study on the neutron irradiation damage of CLAM steel

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China low activation martensitic (CLAM) steel, one of the three main reduced activation ferritic/martensitic steels (RAFMs) under development in the world, has been selected as the primary structural material of ITER testing blanket material (TBM) in China. It is important to understand the neutron irradiation effects of CLAM steel, especially in an environment with high energy and high dose neutrons, as which is much more close to the service condition in future fusion energy systems. A series of neutron irradiation experiments of CLAM steel have been carried out in China and abroad to promote the application process of CLAM steel. In this presentation, current status of the study on the neutron irradiation damage of CLAM steel will be introduced, in which the results from fission and spallation neutron irradiation are included and discussed. Based on the understanding of the application requirements and current results, the R&D strategy of the study on the neutron irradiation damage of CLAM steel will also be proposed to contribute to its industrial applications in future nuclear energy systems.
China low activation martensitic (CLAM) steel has been selected as the primary structure material of FDS series PbLi blankets for fusion reactors, CN helium cooled ceramic breeder (HCCB) test blanket module (TBM) for ITER and the blanket of other future fusion reactors. Tantalum (Ta) is the essential element for reduced activation ferritic/martensitic (RAFM) steels, and the effect of Ta content is one of the key issues for RAFM steels to be finally applied to fusion reactor. To gain the optimal composition of tantalum for CLAM steel, series works had been done by FDS Team. In this paper, the effect of Ta content (four ingots with different Ta contents of 0.027 wt%, 0.078 wt%, 0.15 wt% and 0.18 wt%, respectively) on the tensile, impact, creep and fatigue properties of CLAM steel were presented. The ingots with Ta content of 0.027% and 0.18% had a higher tensile strength and fatigue life, though the difference of those between the four ingots was slight. The ingots with Ta content of 0.078% and 0.18% had the lower ductile brittle transition temperature (DBTT) value. The elevation of Ta content caused an obvious increase in creep rupture time of one order of magnitude. Meanwhile the creep rate and fatigue softening rate were all inhibited with the increase of Ta content. The grain distributions, precipitates and dislocation evolution combined with theoretical calculation were analyzed. With Ta content increasing, the content of Ta-rich MX particles increased and the Cr-rich M23C6 carbides reversed. Meanwhile, the grain size became finer. For the ingot with Ta content of 0.027%, the effect of precipitation strengthening of Cr-rich M23C6 carbides was more obvious than fine-grain strengthening. And the MX particles were beneficial to hindering the movement of dislocations which could decrease the creep rate and fatigue softening rate.
Status and planning of ITER material activation experiments at JET


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Activities under the EUROfusion work package (WP) JET3 programme have been established to enable the technological exploitation of the planned JET experiments over the next few years, which culminates in a D-T experimental campaign, DTE-2. In the areas of nuclear technology and nuclear safety the programme offers a unique opportunity to provide experimental data that is relevant to ITER. The key purpose of the collected data will be to support benchmarking and validation activities relating to neutronics and activation codes, and associated nuclear data, that are used to predict the nuclear behavior of ITER component and materials, during and after operations. This paper details the status and key issues of the ongoing ACT sub-project under WP JET3, which aims to take advantage of the large 14 MeV neutron fluence expected during JET DTE2 to irradiate samples of real ITER materials used in the manufacturing of the main in-vessel tokamak components. The materials considered, with specified minor elemental impurity levels, include: Nb3Sn, SS316L steels from a range of manufacturers, SS304L, Alloy 660, Be, W, CuCrZr, OF-Cu, XM-19, Al bronze, Nb3Sn, NbTi and EUROFER. The activities include provision for measurement of nuclide activities for each material and comparison against the predicted quantities through calculation with the FISPACT-II inventory code. Included here are key pre-analysis results for the selected ITER irradiation samples, and corresponding optimization of diagnostic foils (Ti, Mn, Co, Ni, Y, Fe, Co, Sc, Ta) that will be irradiated at selected positions inside JET irradiation stations in order to determine the neutron spectrum. Preliminary experimental activation results through recent JET DD operations are discussed. This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053 and from the RCUK Energy Programme [EP/I501045].
EU DEMO safety and operating requirements. Issues and possible solutions

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The preliminary safety and operating design requirements are being defined aiming at obtaining the license for construction with a relatively large operational domain to assure an easy control and adequate availability of DEMO. The DEMO design approach is being organized, by taking into account the Nuclear Power Plant experience and the lessons learnt from ITER and GEN IV. Outstanding challenges remain in several areas with potentially large gaps beyond ITER that need to be overcome and require a pragmatic approach. Here the focus is directed mainly to evaluate and improve the readiness of technical solutions through a dedicated R&D program. Integrated Plant Design Assessments are important since the early phase to provide an integration capability of various engineering and operational options with the relevant impact on interfacing systems, e.g. primary heat transfer, electrical power supply, layout and remote maintenance. The overall DEMO plant design has to be strongly safety and operation-balance of plant oriented. The paper describes a few leading aspects of safety and balance of plant that require early attention and a continuous reanalysis at any significant design change, e.g.: (i) safety provisions required by the coolant options, including the protection and mitigation features following an in-vessel or out-of-vessel loss of coolant accident; (ii) tritium inventory limit control considering the substantial throughput of fueling; (iii) the conditions for a plasma shutdown, (iv) the pulsed operation and the relevant interfaces with the grid and with the main BoP systems; (v) the tokamak building layout that has to accommodate RM and to meet layout and environmental conditions criteria. Any effort to reduce the complexity of a Fusion Power Reactor through simplification and rationalization of the design and operation will translate into beneficial returns on safety and on operation and for a higher flexibility with respect to the integration of sub-system options.
P1.198

Gaps analysis for the DEMO safety research and development

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Abstract: A fusion DEMO reactor, like other advanced nuclear energy systems, must satisfy a range of goals including a high level of public and worker safety, low environmental impact, high availability, a closed fuel cycle, and the potential to be economically competitive. It is well known that the experience of the ITER project will facilitate DEMO programs in developing a safety approach and safety design, performing safety analyses under the scrutiny of a nuclear regulator, ensuring reactor availability, managing radioactive wastes, and conducting economic assessments. However, there are still huge scientific and technological gaps between the current ITER and any DEMO reactors. In this work, the international efforts for fusion safety research and development towards DEMO will be summarized following lessons learned from ITER. The main scientific and technological challenges, particularly considering the differences between ITER and DEMO, will be presented with the views not only from the fusion energy development but the development of other advanced nuclear energy systems in particular Generation-IV fission reactor. Moreover, the potential future role of International Energy Agency (IEA) implementing agreement (IA) on a co-operative program on Environmental, Safety and Economic aspects of Fusion Power (ESEFP) will also be addressed in leading the DEMO safety R&D. Keywords: DEMO, Safety Gaps, IEA ESEFP IA
Environmental safety assessment for fusion reactor due to gaseous tritium release

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Environment assessment of large inventory tritium for fusion devices is an important issue before fusion energy commercially used. Different with other radioactive substance, tritium has particular processes of atmosphere dispersion, dry & wet deposition, oxidation in air & soil, reemission, transfer among the soil, plants, animals and human beings. In our previous work, a virtual point source method was utilized for tritium reemission modeling, coupled with dynamic method of the tritium migration in biologic chain to process the accident events. In this contribution, we evaluated the environmental impact and determined the dose sources on the time scale after different fusion accidental release under many conditions. The proportion of intake dose through different ways in the total dose has also been calculated. Besides, according to the principle of consistency between released tritium amounts and limit dose, the maximum tritium release amount has been re-evaluated for different kinds of fusion accidents. The final discussion showed the influence of meteorological condition on the public dose under both normal and accident events. And relevant tritium emergency measures were preliminary proposed.
Radiological impact mitigation of waste coming from the European Fusion Reactor DCLL DEMO

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In large fusion machines, as the foreseen DEMO, the high energy neutrons produced will cause the transmutation of the interacting materials which become a source of radioactive waste. One of the main presuppositions for the global interest in nuclear fusion is that it should be cleaner and safer comparing with traditional nuclear technology. This implies, among other considerations, that the radioactive waste produced in a fusion power plant is expected to be categorized as “low level waste” after no more than 100 year since the shutdown. Besides the main constituents of a material that could transmute but which presence is essential, the impurities, either naturally occurring or originated from the manufacturing process, often give rise to significant additional activation compared to the base material. Thus, once identified the elements generating the dominant activation products in each material of the DCLL (Dual Coolant Lithium-Lead) design option for the European DEMO, the objective of the study has been to determine the impurity limits for such materials in order to minimize the radiological impact of the waste produced and taking into account the industrial viability of the impurity reduction. For both the original compositions and the revised ones (i.e. compositions with reduced amount of impurities suggested to mitigate the waste impact) activation calculations have been performed. Hence, total beta-gamma activity, alpha activity, specific activity for different nuclides, decay heat and surface gamma dose rate have been analysed with reference to the IAEA standards for waste classification and to the specific regulations of the Spanish facility “El Cabrill” for waste disposal. The analyses have entailed the use of the transport Monte Carlo code MCNP5, the inventory code ACAB, and the nuclear data libraries JEFF3.1.1 and EAF2007 for transport and activation respectively.
Activation and decay heat analysis of the European DEMO blanket concepts

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Demonstrating tritium self-sufficiency is an important goal of the European tokamak DEMOnstration reactor developed within the Power Plant Physics and Technology (PPPT) EUROfusion programme. Currently four breeder blanket concepts are being considered: the Helium Cooled Pebble Bed (HCPB), Helium Cooled Lithium-Lead (HCLL), Dual Cooled Lithium-Lead (DCLL) and Water Cooled Lithium-Lead (WCLL). The differences in materials and construction of the four breeder blanket concepts leads to differing nuclear responses under neutron irradiation. This is of particular importance in safety analyses, such as the modelling of loss of coolant accidents, as it affects the blanket’s decay heat and nuclide inventory. This paper presents and discusses analysis performed for each of the blanket concepts to ascertain the decay heat and nuclide inventory for both the entire reactor and individual blanket modules. It was found that the total decay heat at short decay times for the HCLL concept (17.5 MW at 1s) was between 17-22% lower than the HCPB, WCLL and DCLL At longer decay times (~100 years) it was found that the DCLL and WCLL blankets had decay heats in the region of 2-3 orders of magnitude above the HCPB and HCLL blankets. Although the majority of the dominant active nuclides in the tungsten ($^{187}$W and $^{185}$W) and Eurofer (structural steel) ($^{55}$Fe and $^{51}$Cr) were similar between the blanket concepts some differences were noted. For example, longer decay times $^{121}$Sn was one of the dominant nuclides for Eurofer in the HCPB concept; however does not appear in the top ten dominant nuclides for the HCLL concept. The differences in dominant nuclides are discussed in the context of neutron spectra and material compositions.
A comparison study of shutdown dose rate using R2S and D1S method in K-DEMO

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The pre-conceptual design concept on the Korean fusion demonstration reactor (K-DEMO) has been studied in Korea since 2012. In the fusion reactor, neutrons produced from fusion reactions cause activation of fusion reactor devices. For the safety of fusion devices and workers during operation and maintenance, it is important to calculate activation and to evaluate shutdown dose rate (SDR). In this study, SDR for K-DEMO global model was evaluated and compared with two different computational methods called rigorous 2-step (R2S) method and direct 1-step (D1S) method. In the R2S method, cell-wised activation calculation was performed by coupling transport code MCNP and activation code FISPACT. On the other hand, in the D1S method, neutron and delayed gamma transport calculations were simultaneously performed using point-wised MCNP calculations. The SDR was calculated in the four positions of the K-DEMO global model, two inside the vacuum vessel and two outside, at 7 cooling times. The results show that SDR results using D1S method were slightly higher comparing to the results using R2S method. This tendency was higher inside the vacuum vessel than outside. The detailed comparison and explanation is presented in this paper. These results will be able to be utilized to establish further detailed activation analysis procedures for designing maintenance and decommissioning schemes of K-DEMO.
Neutron activation in heat transfer systems of nuclear fusion devices

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Coolant activation is important concern for nuclear fusion devices, where water is being used in heat transfer systems. Production of nitrogen-16 isotope is one of the main hazards in such systems and should be taken with care. In this work, the examination of the neutron activation in water cooling systems, that might be used in future fusion devices, was carried out. Primary heat transfer systems for vacuum vessel, first wall/blankets and divertor, which might appear in ITER and DEMO projects, were examined, focusing on components that are relatively close to the neutron source. Assessment of activity and decay heat of water and functional materials was performed with respect to different localizations and structures. Contact dose rate and dose rate distribution in vicinity of 1 meter were also estimated. In addition, the possible influence of activated corrosion products in coolant was investigated and dominant radionuclides were identified. Calculations were made with few assumptions: flow rate of coolant is steady and uniform; activated coolant does not mix in other cooling systems; coolant is being reused for full fusion device operation period. Irradiation was simulated with activation system program FISPACT. Obtained results showed good agreement with other works in the available literature.
Methodology of the Source Term estimation for DEMO Reactor

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The problem of Source Term qualification is one of the most important topics in order to predict possible releases of the Activation Products (APs) and tritium from the DEMO Fusion reactor. The prevention of any possible consequence, which can affect the environment and the population, is the mission of Fusion technology. In the frame of the EUROfusion Work Package of Safety Analyses and Environment (WPSAE) a methodology to scale and to evaluate the source term provided by ITER and a Conceptual Study of Commercial Fusion Power Plant (PPCS) sources has been studied. This paper refers to the activity currently done for the DEMO source terms assessment and the preliminary results obtained in the ongoing activity. During activities in the task, the methodology was developed for prediction of the tritium and APs concentration. The methodology is explained in details useful for prediction of the tritium and APs concentration in Vacuum Vessel (VV) and in the Breeding Blanket (BB) starting from the DEMO current design data and the inventories assumed in ITER, PPCS and SEAFP programs. These results refer to the Helium Cooled Lead Liquid (HCLL) and the Helium Cooled Pebble Bed (HCPB) concepts. This approach is based on the foundations, set in the fission technology safety analysis of the Design Basis Accidents (DBA), Design Extension Conditions (DEC) and Beyond Design Basis Accident (BDBA).
Development and identification of detritiation techniques for DEMO radioactive waste management

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In frame of project Eurofusion, WPSAE (safety and environment) were reviewed existing detritiation technique for different material types and identified techniques for further development for short-term reuse, long-term reuse, recycling and disposal. Moreover criteria for assessment were proposed and technique were described. The most efficient treatment technique for different group of material types proposed for DEMO (DEMO is the successor of the international fusion experiment ITER) were established.
Disposal Procedure for contaminated surface of tritium handling facility in the decommissioning operation

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After the tritium handling operation, it is an important issues to take an appropriate disposal method of tritium handling facility contaminated with tritium. In Kyushu University, according to the relocation program to the new campus, decommissioning operation of tritium handling facility located in the former campus had been performed. This handling facility made of concrete was used for accelerator experiments with several tritium targets (370 GBq/target) from 1961 to around 1980. However, detailed experimental conditions are not found. Prior to launch the total surveillance of handling facility, pre-measurement was performed for 325 surface smear points and 10 core sample points for depth profile on concrete floor and wall to determine the way of total surveillance. Remarkable amount of tritium contamination was observed in approximately 80\% of total surface smear points. Depth profile of tritium concentration obtained from core sample showed that most of highest contamination points existed in the point face the operation room but some of the highest points existed in the deeper point. This may be explained that during the experimental period, tritium released or leaked into the operation room penetrates into the concrete wall with concentration gradient. After shutdown of the accelerator experiments, tritium exists in the concrete wall transferred to the room air during long time after shutdown because no tritium exists in the room air. As the results, highest tritium concentration point was observed at the deeper point. Attention should be paid not only to measure tritium on the surface, but also to tritium exists in the inside of wall and detecting method in the decommissioning of tritium handling facility. In the total surveillance, we took a scraping method from inner surface divided into 1m\textsuperscript{2} of area and re-scraping was performed when significant contamination was detected in the first surface measurement.
P1.207

Environmental impact of nuclear fusion biomass gasification plant

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In previous studies, the authors proposed a novel nuclear fusion biomass gasification plant concept as an alternative to conventional nuclear fusion power plants. This gasification plant concept utilizes the heat from fusion blanket to convert biomass into synthetic gas (H₂ + CO), and then convert it into liquid fuels, e.g. methanol or diesel. Through this nuclear fusion gasification plant concept, the authors believe that a highly efficient, sustainable, and environmentally friendly liquid fuel production could be achieved. In this study, through life cycle assessment (LCA), the authors quantitatively assessed and compared the environmental impacts of this plant concept for the first time against conventional gasification plants. Assessed impact categories were: acidification, ecotoxicity, eutrophication, global warming, human health – carcinogenic, human health – non-carcinogenic, ozone depletion, photochemical ozone formation, resource depletion and respiratory effects. The subject process is a methanol production from wood chip via synthetic gas, at the process rate of 2,000 odt/day. Fluidized bed gasifiers were assumed at the plants. The system boundary of the LCA was set to cradle-to-gate of the methanol, with the functional unit of kg-methanol production. TRACI 2.1 method was adopted for the impact assessment, and Ecoinvent Version 3 was used for foreground data. LCA results indicated that the nuclear fusion biomass gasification plant would reduce the GHG emissions by 2.26 kg CO₂-eq per 1 kg of methanol production, or as large as 43%, when compared to the conventional gasification plants. Therefore, it was concluded that this novel nuclear fusion biomass gasification plant would have a great potential as an alternative, sustainable source of liquid fuel in the future.
3D effects of tokamak ferromagnetic core on magnetic field

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A number of tokamaks, including the largest operating one, Joint European Torus (JET), has ferromagnetic core installed in their plasma current drive system. Moreover, some auxiliary systems, such as magnetic shielding of neutral beam injection (NBI) system, or iron inserts for toroidal field ripple mitigation, consist of non-negligible amount of ferromagnetic material as well. Besides the intended favorable effect of these materials on a specific magnetic field, there might be also unintended detrimental effects on the other magnetic fields. These need to be evaluated prior the installation of the ferromagnetic material and taken into account in the interpretation of the magnetic measurements and modelling. This contribution presents a model used to simulate the effects of ferromagnetic material with general-geometry on distributions of magnetic field in the vicinity. It is based on integral method approach, which has the advantage in its computational simplicity. The model has been benchmarked on two tokamaks with ferromagnetic core (GOLEM and STOR-M) and then applied to characterize the effect of the core on the distribution of poloidal magnetic fields on these devices. Lastly, the effects of JET ferromagnetic core on the field induced by error field correction coils is evaluated by the presented model and discussed.
Modelling and analysis of the JET EP2 neutral beam FEID curved end plate

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Neutral beam injection systems have proved themselves as the most effective form of auxiliary heating in tokamak plasmas. In positive ion based systems once the beam is neutralised there are many residual ion components which must be intercepted by suitable ion dumps. A particular challenge for ion dump design occurs when the dump must be placed close to a focus point as is the case for the curved end plate of the JET NBI full energy ion dump. Molecular ion species though of low power are focused at this place. As part of the EP2 upgrade to increase neutral beam power and duration, the ion source configuration was changed from Supercusp 130kV/60A configuration to Chequerboard 125kV/65A. This allowed for significant increase in neutral beam power but also lead to a fourfold increase in molecular residual ions. The curved FEID end plate was re-designed as an actively cooled element using swirl tubes. Following a failure of this plate in 2014 additional analysis was carried out to determine likely causes of the failure and to improve its performance. This paper describes enhanced modelling of the power loading, improvements to the power handling capabilities and additional features to improve fatigue life. Monte-carlo simulations of each of the nine residual ion components which are intercepted by the plate shows a peak power density of 25MW/m² and compares well with recently installed fast thermocouple measurements. Analytical calculations and simulations with the Charged Particle Optics (CPO) code are used to investigate the potential for movement of the residual ion focus due to space charge neutralisation effects. Cooling performance is significantly enhanced by improved water channel flow which is both modelled and confirmed by experiment. Fatigue life, calculated from ANSYS modelling is improved using a slot arrangement to relieve stresses created from focussed heat load distribution.
Technical rehearsal of DT operation at JET

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The final phase of the JET Programme in Support of ITER plans to operate with 100% Tritium (TT) followed by Deuterium-Tritium (DT) operation, to help minimise risks and delays in the execution of the ITER Research Plan and the achievement of Q~10. Additional technical requirements (compared to Deuterium operation) are needed to allow operation with Tritium gas, a high DT neutron flux and neutron activation. These additional requirements include (1) the supply with Tritium of both neutral beam boxes and five new gas introduction modules, (2) the access restrictions to personnel to key operational areas and computer systems for prolonged duration, (3) the depression and oxygen depletion to 15% of the torus hall, (4) the daily regeneration of the divertor and neutral beam cryo-pumps and (5) additional operational procedures and experiment preparations for the accounting of the tritium inventory and neutron activation. As part of the preparation for the TT and DT campaigns at JET, an 8-week technical rehearsal of the procedures and systems to be used in these campaigns is planned in 2016. The aim is to characterise the NBI performance with a Tritium-like gas feed configuration and to rehearse the operation of the technical systems, to test operational procedures and most importantly to gain experience and make recommendations for the preparation of future Tritium campaigns on JET. This contribution will give an overview of the preparation carried out for the DT rehearsal and the results obtained and lessons learned from the technical rehearsal itself. 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.
Neutronics benchmark experiments are conducted at JET for validating the neutronics codes and tools used in ITER nuclear analyses to predict quantities such as the neutron flux along streaming paths and dose rates at the shutdown due to activated components. In particular, in the frame of subproject NEXP of JET-3 program, several activities are performed within EUROfusion Consortium devoted to the preparation of neutronics experiments for the future Deuterium-Tritium operations (DTE-2 campaign). During plasma operations, neutron fluence and dose measurements will be performed using thermoluminescent dosimeters (TLDs) and activation foils located in several positions inside and outside the Torus Hall. At the shutdown, decay gamma dose rates will be measured using passive and active dosimeters installed inside and outside JET vessel. Decay gamma dose in-vessel measurements will be performed with TLDs and high-sensitive, low activation, spherical ionization chambers will be used to measure the dose rate versus time after irradiation in two ex-vessel positions on the side-port of Octant 1 close to radial neutron camera and in Octant 2 on the top of ITER-like antenna. The results of the measurements will be compared with three-dimensional calculations carried-out with MCNP5 and MCNP6 Monte Carlo Codes as well as with ADVANTG hybrid code. Shutdown dose rate measurements will be used to validate recent versions of three-dimensional MCNP-based Rigorous-Two Steps and Direct-One Step European tools used in ITER analyses. The experimental assembly has been accurately designed so far and careful analyses have been performed for the selection of the detectors as well as the experimental positions and operative conditions. Measurements and analyses are in progress in the current Deuterium-Deuterium (DD) campaign. This work is devoted to present the status-of-art of neutronics experiments including selection and detectors’ assembly, irradiation tests performed in neutron and gamma facilities, calibration, pre-analyses and recent results during DD operations.
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. This tokamak should be capable of hosting scaled experiments integrating most of the possible aspects of the DEMO power and particle exhaust. DTT should retain the possibility to test different divertor magnetic configurations, liquid metal divertor targets, and other possible solutions for the power exhaust problem. In this paper we investigate the feasibility and the costs of conventional and alternative magnetic configurations on DTT. We have developed Single Null, SnowFlake, Quasi SnowFlake 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). The Nb3Sn central solenoid (CS) is capable to operate at 13.2 T while the poloidal field (PF) coils work in a not-challenging range of parameters for the superconducting NbTi material. The alternative configurations are compared in terms of various parameters such as plasma current, plasma volume, the flat-top magnetic flux swing and PF current request.

DTT Device: Conceptual design of the superconducting magnet system

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In the European Fusion Roadmap, one of the main challenges to be faced is the mitigation of the risk due to the impossibility of directly extrapolate to DEMO the divertor solution adopted in ITER, due to the expected very large loads. Thus a satellite experimental facility oriented toward the exploration of robust divertor solutions for power and particles exhaust and to the study of plasma-material interaction scaled to long pulse operation, is currently being designed. Clearly, this kind of experiment presents challenging design requirements, due to the extreme operation conditions which shall be as representative as possible of the DEMO ones, but with much smaller dimensions and lower costs. Our team has performed a feasibility study for a fully superconducting magnet system of a compact tokamak reactor, in the framework of the activities carried out in Europe for the Divertor Tokamak Test (DTT) facility project. In the conceptual design presented in this paper, the magnet system is based on Cable-In-Conduit Conductors, adopting, whenever possible, the most recent developments in the field. It consists of 20 Toroidal Field, 6 Poloidal Field and 6 Central Solenoid module coils. The proposal is based on preliminary reference parameters such as plasma major and minor radii, magnetic field on plasma axis, plasma current, inductive flux etc., which should assure plasma scenarios suitable to investigate reliable power exhaust handling solutions. The main aspects driving the magnets design, from mechanical to thermo-hydraulic analyses, are here presented and discussed.
The DTT device: systems for heating

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The proposed Divertor Test Tokamak, DTT, aims at studying power exhaust and divertor load in an integrated plasma scenario. Additional heating systems have the task to provide heating to reach a reactor relevant power flow in the SOL and guarantee the necessary PSEP/R together adequate plasma performances. About 40 MW of heating power are foreseen to have $\text{PSEP/R} \geq 15 \text{ MW/m}$. A mix of the three heating systems presently proposed for ITER has been chosen, assuring the necessary flexibility in scenario development. An ECRH system at 170 GHz will provide 10 MW at plasma for several tasks, such as: bulk electron heating to bring the plasma in the high confinement regime, current profile tailoring by localized CD, avoidance of impurity accumulation, MHD control and current ramp up and ramp down assistance. Together with the EC system, 15MW of ICRH (in the range 60-90MHz) will provide the remaining bulk plasma heating power, on both electrons and ions. ICRH, in minority scheme, will produce fast ions, with an isotropic perpendicular distribution, allowing the study of fast particle driven instabilities like alphas in D-T burning plasmas. The heating schemes foreseen in DTT are $^3$He and H minority as well as Deuterium 2$^{nd}$ harmonic. The addition of 15 MW of NBI, later in the project, could provide a mainly isotropic parallel fast ion distribution to simulate the alpha heating scheme of a reactor. The NBI primary aim is to support plasma heating during the flat top phase when the need of central power deposition and the minimization of the shine-through risk suggest a beam energy around 300 keV. In the first phase of the DTT project the available power will be at least 25 MW, to be increased during the lifetime of the machine.
on behalf of the EUROfusion WPDTT2 team & the DTT report contributors Within the frame of
the DTT program, included in the EuroFusion roadmap, the design of a new Tokamak dedicated
to tackle the Power Exhaust problem as an integrated bulk and edge plasma problem has
been developed. The main guidelines used to work out the machine parameters will be shortly
illustrated. To allow the machine flexibility in withstanding power exhaust as large as possible,
a set of small internal coils will be installed in order to study many of the advanced magnetic
configuration presently evaluated. Divertor magnetic configurations with different geometries and
materials (including liquid metals) will be reported by several examples. The machine is foreseen
to start the operations with a standard divertor configuration of actively cooled W. The first wall
will consist of removable panels with a coating of W; its temperature will be actively controlled
aiming to operate up to a temperature of around 300° ÷ 400° C. The main expected machine
performances have been explored by using different numerical codes. In particular the integrated
plasma bulk and edge features have been studied by using the code COREDIV, also performing
runs with different impurities at different concentrations.
The DTT device: first wall, vacuum vessel and cryostat structure

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This paper describes the activity addressed to the conceptual design of the first wall and the main containment structures of DTT device, which will be broadly presented in the invited talk “Design and definition of a Divertor TOKAMAK Test facility”. The work moved from the geometrical constraints imposed by the desired plasma shape and the configuration needed for the magnetic coils. Many other design constraints have been taken into account such as remote maintainability, space reservations for diagnostic and heating equipment, etc. The basic vessel design resulted in an all-welded single-wall toroidal structure made of 18 sectors. Proper supports have been designed for the first-wall, which was conveniently segmented in view of remote maintenance. This provisional model allowed evaluating the electromagnetic loads on the metallic structure of the vacuum vessel resulting from the current quench due to a plasma disruption. After a FEA mechanical assessment, which was conducted according to ASME code, INCONEL 625 stainless steel has been provisionally selected as reference material for vacuum vessel. 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 will hold the vacuum vessel and the magnets.
The DTT device: general layout

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The DTT (Divertor Test Tokamak) is a new facility conceived in the frame of EUROfusion roadmap with the aim to assess and possibly integrate all the relevant physics and technology divertor issues. The general project is presented in another paper of this conference [1] and with more details in [2]. The general project includes the analysis of the site requirements from several points of view; among other alternatives the ENEA Frascati Research Center (FRC) has been indicated on the basis of technical-scientific, organizational and economics considerations. FRC is well suited from this point of view. Since 1960, FRC hosts most of the Italian fusion research. Presently the FTU machine is in operation at FRC. For the DTT plant requirements it will be possible to adapt the complex FTU buildings except the DTT hall and the cryoplant. The DTT hall will be an extension of the present FTU hall. The machine would be preassembled in a modular way inside the present FTU hall, which, on a longer time scale, should host the NBI injector. The dimensions of the new hall are 30x20x28 m on three levels. On the lowest one, the cold boxes for the electrical connection of the superconductive coils will be placed while in the intermediate level the diagnostic using the bottom ports will be arranged. The third level starts at the cryostat bottom and will host all the additional heating system and the diagnostics. The machine is particularly demanding in terms of power supplies and the grid requires an extension of the 150 kV line. Discussions are in progress with the operators for energy transmission. The tunnel solution is recommended to prevent possible environmental impact. [1] R. Albanese et Al. “Design and definition of a DTT facility”, this SOFT 2016 Conference [2] http://fsn-fusphy.frascati.enea.it/DTT_ProjectProposal_July2015.pdf
The DTT device: power supply and electrical distribution system

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The power supplies (PSs) of the DTT proposal, as presented in the talk “Design and definition of a Divertor Tokamak Test facility” invited at this conference, have to feed:

- 6 central solenoid (CS) and 6 poloidal field (PF) superconducting coils, with currents up to 25 kA.
- 18 toroidal field (TF) superconducting coils, with a current up to 50 kA. Some fast plasma control coils, including at least 2 internal coils for vertical stabilization. The electron (ECRH) and ion (ICRH) cyclotron additional heating systems, for about 25 MW delivered to the plasma.
- All auxiliary systems and services for more than 100 MVA.

An upgrade of the heating system, able to deliver to the plasma further 20 MW, especially by a neutral beam injector (NBI).

The CS and PF PS circuits include a 4-quadrant 12-pulse AC/DC converter in series to a quench protection circuit (QPC) and, in most cases, a switching network unit (SNU). The voltages and currents to be provided by the converters were estimated applying the reference scenarios to a model of the PF circuits, taking into account the mutual couplings and the SNU contributions.

The independent evaluation of the electrical requirements of each PS system led to the definition of active, reactive and apparent power scenarios. Due to the pulsed PSs (serving CS, PF, ECRH, ICRH, NBI), the 100-MVA continuous load can reach 350 MVA with a duty cycle of 100s/3600s. The feasibility of these scenarios at the ENEA Center in Frascati was verified, including a compensation of the power factor up to 0.9, also with the support of the Italian transmission grid operator. The solution identified to supply all the facility directly from the national grid requires a new 150 kV line specifically for DTT and a new substation with two 150kV/36kV transformers inside the ENEA Center.
The experimental facility THALLIUM (Test HAmmer in Lead LithIUM) was designed to experimentally validate the RELAP5-3D code simulations of the pressure wave propagation in the HCLL TBM due to In-box LOCA. THALLIUM, which reproduces the geometry of the LLE loop of the HCLL TBM, was installed at the ENEA Brasimone Research Centre to support the accidental analysis of this type of test blanket module. Within the framework of F4E-FPA-372, an experimental campaign was carried out in ITER relevant conditions. The experiments simulated a pipe rupture in a cooling plate of the HCLL TBM. The main objective of our campaign was to study the release of high pressure helium in the LLE. Particularly, one of the fundamental phenomena to be observed is the pressure wave trend in the two pipes that reproduce the pipe forest of ITER. Three parameters were varied during the experiments: the set pressure of the rupture disk, the helium injection mass flow rate and the opening pressure of the relief valve. The second objective of this campaign was to validate the system code RELAP5-3D with our data. Furthermore, an additional task was to test new instrumentation and, in particular, pressure meter with acquisition time of 1kHz and a level meter in the expansion tank. In the pipe forest mock-up, the pressure wave displayed three distinct steps of growth and reached the maximum value in about 4 seconds. These three increases were likely caused by the arrival of the incoming wave, the wave reflected by the isolation valve and the He that reaches the measurement point. The results show that the behaviour of the system is strongly influenced by the isolation valve closure and by the opening pressure of the relief valve. The validation of the code on the basis of the experimental results are being carried out.
He-FUS3 experimental campaign outcomes and RELAP5-3D analysis

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The 1st Specific Grant of the Framework Partnership Agreement 372 deals with experimental activities in support of the Conceptual Design of HCLL and HCPB Test Blanket Systems. Service-2 is focused on thermal-hydraulic tests with high pressure Helium for validation and benchmarking of suitable dedicated numerical tools. In this frame, an extensive experimental campaign has been executed in He-FUS3 (European helium cooled blanket integral test) facility designed and realized at ENEA Brasimone Research Center, to test the thermal-mechanical properties of prototypical module assemblies of the ITER and DEMO reactor. The actual facility has been upgraded with an elevated performances turbocirculator and an additional water cooling system that integrates the pre-existent Air Cooler. In addition, a dedicated Test Sections located in the loop hot zone has been settle down with the objective to investigate safety relevant transient conditions of an “In-TBM” LOCA event. Experiments have been conducted for a wide range of HCS relevant operating conditions, in order to investigate the whole facility T/H performances (cold and hot conditions) with special interest in the turbocirculator operating region assessment. Moreover, incidental tests representative of LOFA, “In-TBM”/“CVCS Area” LOCAs scenarios have been included. The main experimental outcomes, herein reported and discussed, have also provided a valuable data base for T/H system codes validation, thus allowing the implementation of a newly RELAP5-3D©©© numerical model of the actual He-FUS3 layout. A post-test numerical analysis was performed in order to evaluate the model consistency and validate the code capability to adequately reproduce and predict the system behaviour. Code results have been compared with experimental data and the main issues related to the modelling capabilities have been addressed.
Experimental and RELAP5-3D results on IELLLO (Integrated European Lead Lithium LOop) operation

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The experimental facility IELLLO (Integrated European Lead Lithium LOop) was designed and installed at the ENEA Brasimone Research Centre to support the design of the HCLL TBM. This work presents the results of the experimental campaign carried out within the framework of F4E-FPA-372 and which had three main objectives. First, to produce new experimental data for flowing LLE (Lead-Lithium Eutectic) for an analysis of the loop and the characterisation of its main components. Then, to evaluate performances of commercial instrumentation as available instrumentation is not designed for use in LLE. Lastly, to use the data for validation of the model developed with the system code RELAP5-3D. An additional objective was to simulate the emergency drainage in order to evaluate the time required to complete the process. The data collected could prove helpful to analyse the behaviour of the LLE loop of ITER and DEMO in accidental conditions. The results show that the regenerative countercurrent heat exchanger has an efficiency ranging from 70 to 85%, mainly depending on the LLE mass flow rate. We verified that the air cooler has the capability to keep the cold part of the loop at 350°C, even in the most demanding situation (700 rpm and maximum temperature of the hot part). The instrumentation tested showed essentially good accuracy, with the exception of the turbine flow meter. Nevertheless, specific limitations in the upper operative temperatures were found for the LLE direct contact pressure transducer. We found a good fit between the experimental results and the associated RELAP5-3D simulations. Care should be taken during the simulation of the draining process to avoid convergence issues.
Particle density uniformity based global weight window generator in monte carlo particle transport simulation

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Due to the complexity of fusion reactors on geometry and neutron physics, the Monte Carlo (MC) methods have been broadly adopted in fusion nuclear design and analysis. But for calculations that require obtaining a detailed global flux map, such as the shutdown dose rate analysis, analog MC simulations usually cost a prohibitive long run time. To make such analysis computational practicable, it is necessary to adopt an efficient global variance reduction (GVR) method. This paper proposed a new mesh weight window based GVR method, named Global Weight Window Generator (GWWG). For each weight window cell, this method calculates its importance as the expected contribution to the particle density uniformity generated by a unit weight particle entering this cell. This contribution is calculated in a way trying to reach a balance between penetrating deeper by splitting and simulating more source particles per unit time. It also exploits an efficient and fully automatic iteration scheme to speed up the weight window generation. The development of the GWWG method is based on the SuperMC code, which is a general, intelligent, accurate and precise simulation software system for the nuclear design and radiation safety evaluation. To validate the performance of the GWWG method, series of tests have been performed with the ITER benchmark, ITER Alite and the ITER Clite model, calculating the neutron flux over a mesh tally covering the entire reactor. All the tests have showed a substantial increase in computing efficiency compared with the analog case. The highest speedup in the MC figure of merit, ~249 times, is achieved with the ITER Alite model. These calculations demonstrate the ability of the GWWG method to greatly enhance the efficiency of global flux map simulation of complex models. Keywords: Monte Carlo; SuperMC; Global variance reduction
Advanced Capabilities of Monte Carlo Program SuperMC for Fusion Application

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Great challenges exist in real fusion engineering projects for the current Monte Carlo (MC) methods including the calculation modeling of complex geometries, simulation of deep penetration problem, slow convergence of complex calculation, lack of experimental validation for new physical features, etc. Several novel and advanced capabilities of the latest version of MC program SuperMC for fusion applications were introduced in this paper. An automatic and intelligent CAD-based modeling function is developed. The latest ITER reference model C-Lite with complex solids and hierarchy structure can be modeled. The output data can be automatically and intelligently visualized by mixing with the input models according to users’ interests. The simulation process and real-time dose can be visualized to test and evaluate the operational or maintenance tasks and assist the supervisors to plan better working activities. Three dimensional domain hybrid MC and discrete ordinates (SN) modeling and transport calculation method with transition region has been developed. Adaptive variance reduction technique for local tally with hybrid MC–deterministic method with weight window smoothing was studied. The optimal spatial subdivision method was employed to enhance the geometry navigation performance. The bounding box algorithm can be specifically customized and applied to accelerate the basic function of calculating the distance to volume boundary. Based on Chebyshev rational approximation method, the activation calculation function was developed. Cloud computing framework makes the calculation and analysis more attractive as a service. SuperMC has been verified and validated by more than 2000 benchmark models and experiments. Series of fusion reactors were employed to validate the comprehensive capability. As the supplementary of validation experiments for fusion applications, an experiment to validate the deep penetration problem of radiation shielding using High Intensity D-T Fusion Neutron Generator (HINEG) which produce 14.1MeV neutrons with \(10^{12} n/s\) beam yield is being particularly conducted.
Comparison of detritiation techniques for purely tritiated metallic waste

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Operation of fusion facilities using deuterium and tritium to fuel the fusion reaction will lead to generation of radioactive waste during operating and decommissioning phases. Most of these wastes are expected to be contaminated with tritium and will require a specific management strategy taking into account the physical and chemical properties of tritium. The reference management strategy for tritiated waste that cannot be accepted directly for its final disposal is a 50-year interim storage enabling tritium to decay (the 50-year interim storage corresponds to 4 tritium radioactive periods i.e. a tritium reduction factor of 16). The most contaminated categories may need to be processed using detritiation techniques in order to reduce the tritium content and tritium outgassing as an alternative or as a complement to an interim storage phase. The detritiation of metallic waste by thermal treatment and melting has been investigated and the main features are described in this paper. Melting offers a higher detritiation factor than thermal treatment. Comparison of different waste management strategies against several independent parameters related to the environment, safety, and technical feasibility has been also performed. For purely tritiated metallic waste made from stainless steel, in any case, performing a detritiation appears to be very attractive, allowing a significant decrease of interim storage duration.
Pre-assembly and dimensional inspection at factory of JT60-SA Cryostat Vessel Body Cylindrical Section

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The superconducting tokamak JT-60SA, aimed to support and complement the ITER experimental programme, is currently being assembled at the JAEA laboratories in Naka (Japan). Within the European contribution, Spain is responsible for providing JT-60SA cryostat. The cryostat is a stainless steel vacuum vessel 14m diameter, 16m height which encloses the tokamak providing the vacuum environment (10-3 Pa). Due to functional purposes, the cryostat was divided into two large assemblies: the Cryostat Base (CB) and Cryostat Vessel Body Cylindrical Section (CVBCS). The CVBCS is presently being manufactured by a Spanish company and it is expected to be delivered in Naka by the middle of 2017. This piece does not include the top lid which must be provided by JAEA. The CVBCS consist of 12 individual sectors made from SS304 plates 34mm thick, weighing up to 175 tonnes, 14m external diameter and 11m height. The sectors are fully welded structures further machined at the connection flanges to get the required tolerances. The CVBCS will be assembled by mechanical connection between the individual sectors. As part of the manufacturing process after the final machining, the dimensional inspection (DI) of every individual sector is carried out by laser tracker (LT) to check the tolerances of the single pieces. Due to the large dimensions of the sectors and their high flexibility, the way to support the pieces resulted very critical for the DI as it was predicted by finite element analyses carried out. Ad hoc jigs were built to support the sector during the measurements in order to avoid distortions. Afterwards, the structure will be pre-assembled at factory for final DI. The paper will summarize the measurement procedure for the dimensional inspection of the individual sectors of CVBCS by LT, the pre-assembly and final DI procedure of the whole CVBCS.
Conceptual design of the JT-60SA pellet launching system

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A conceptual design for a pellet injection system will be worked out, capable to support key missions of the new tokamak device JT-60SA. For exploitations in view of ITER and to resolve key physics and engineering issues for DEMO, several tasks were assigned to this system. Physics investigations aim at operation at high density in ITER and DEMO relevant plasma regime above Greenwald density, power exhaust techniques with radiation layers, particle balance studies, and ELM control and mitigation. The postulated engineering requirement is to quantify pellet actuation on electron density for application within the advanced real-time control scheme by controlling density gradients. Our intended pellet system comprises three major components: pellet source, accelerator and guiding system. The guiding system must be installed inside the torus vessel already under construction, hence still possible launch geometry were pursued first. Three different options have been identified: inboard, outboard and top launch. The first one is most promising with respect to fuelling performance but will impose pellet speed restrictions to about 470 m/s for adequate pellet sizes. Both others offer headroom for significantly higher injection speed but under less favourable physics boundary conditions. In order to evaluate expected performances for all relevant plasma scenarios, detailed modelling efforts for every launch geometry option are in progress. For a suitable pellet source covering all requirements, several options are at hand including commercial providers. For the accelerator, the high speed option up to about 4000 m/s could be covered by a multi stage gas gun. Single stage gas guns and centrifuges can cover the speed range up to about 1000 m/s for the basic work load since both fulfil the requirements for pellet size and speed. Due to a to higher speed precision resulting in less timing jitter, a centrifuge would be better suited for control requirements.
Feasibility study of a flux-gate magnetic field sensor suitable for ITER Neutral Beam Injectors

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The ITER Heating Neutral Beam (HNB) injectors shall be protected from stray magnetic field (several hundreds of mT) produced by the ITER PF coils and plasma current. Such stray field would hamper the production of negative ions, deflect ion trajectories in the accelerator and cause intolerable heat load on neutralizer and beam line components. In order to keep the residual magnetic field below acceptable levels (1 mT in the ion source and accelerator and 0.3 mT in the neutralizer), each injector will be surrounded by Passive Magnetic Shield and by six Active Correction and Compensation Coils (ACCC). The ACCC will be feedback controlled using magnetic field sensors located inside the HNB vessel, an environment subjected to considerable neutron flux (~10⁹ n/cm²/s) during the ITER operation. Therefore, magnetic sensors that are robust, radiation hard, drift-immune and remote-handling compatible are required. Flux-gate magnetic sensors are good candidate, as their active part includes no semiconductor or other radiation-sensitive component, and consists of a ferromagnetic core and two insulated-wire coils. Commercial flux-gate sensors, used for precise measurements of weak magnetic fields, have very good sensitivity, but measurement range below 0.1 mT, and also include on-board electronics. A flux-gate sensor for the ITER HNB has been studied using a numerical model of the magnetic core hysteresis, essential to describe the sensor operation, sensitivity and measurement range. This model indicated that, by suitable choice of the core magnetic properties and gap geometry, the measurement range can be extended by at least 2 orders of magnitude. On this basis, a prototype flux-gate sensor has been realized at Consorzio RFX. Experimental tests carried out so far have confirmed the results of the numerical model and have demonstrated that the measurement range can be increased to ~10 mT with acceptable accuracy and frequency bandwidth.
Realization of a magnetically compensated extraction grid for performance improvement of next generation NBI

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In the multi-beamlet, negative-ion based Heating Neutral Beam (HNB) Injectors presently used in fusion research, arrays of permanent magnets are embedded in the Extraction Grid (EG) for the suppression of the unwanted co-extracted electrons. These magnets cause a significant undesired deflection of the negative ion beamlets, with a typical alternate pattern, matching the orientation of the magnet arrays. As a consequence, a delicate adjustment of the profile of the EG apertures has to be introduced in order to compensate for this deflection by the electrostatic means. During the final design of MITICA, which is the full-scale prototype of the ITER Heating Neutral Beam presently under construction in Padova, a new solution has been developed by Consorzio RFX for the compensation of this undesired deflection. This new approach makes use of an additional set of permanent magnets, called Asymmetric Deflection Compensation Magnets (ADCM), also embedded in the EG, with a series of advantages in terms of performances and versatility with respect to the traditional electrostatic compensation. For this reason, this solution has been adopted as the reference for MITICA, ITER and possibly for DEMO NBI. After a thorough validation of the design by different numerical models, this solution will be experimentally tested for the first time on the Negative Ion Test Stand (NITS) at JAEO Naka Fusion Institute within the framework of a scientific cooperation agreement between JAEO and Consorzio RFX. To this purpose, an EG having an ITER-like profile and compatible with NITS accelerator has been designed and constructed at Consorzio RFX. This paper describes the design solutions adopted for this grid, the construction, and the final assembly procedure of the permanent magnets inside the grid.
Experimental characterization of the MITICA neutralizer gas injection nozzles

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The gas cloud inside the neutralizer of MITICA (Megavolt ITER Injector and Concept Advancement), required to neutralize the negative ion beam, will be created continuously by 20 identical nozzles providing the gas needed for different operation modes. In order to validate the design, one nozzle will be characterized in detail and for a wide range of supply conditions in a dedicated experiment at KIT. This is necessary due to a number of uncertainties in the calculation and prediction of the nozzle behavior. The characterization will comprise two general scenarios: a steady state supply for normal beam operation and a transient test to check the behavior in the case of an accidental interruption of gas supply with the potential risk of a damage of other beamline components. All the experiments on the MITICA gas nozzle will be carried out at the modified TransFlow facility at KIT. In this paper the experiments at KIT to characterize the gas supply nozzles for MITICA in a parametric way over a wide range of conditions are described. The achieved results are then compared with the theoretical predictions already available. Finally, the results from the single nozzle test are used for a scale-up exercise to full MITICA scale. While the current design comprises 20 identical nozzles, arranged in 4 parallel lines of 5 nozzles in series, there is a potential issue of flow maldistribution and its consequences on the system performance. This issue could result from the serial nozzle arrangement: the last in line sees a significantly lower mass flow. With the achieved parametric characterization of one nozzle, also an analytical prediction of the differences between the individual nozzles in MITICA can be made. This result can validate the design of the nozzles and allows an optimization of the operation conditions of MITICA.
Final design of acceleration grid power supply conversion system of MITICA neutral beam injector

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The Acceleration Grid Power Supply supplies the acceleration grids of the MITICA experiment, the full scale prototype of the ITER Neutral Beam Injector under construction in Padua (Italy) to tackle the technical challenges and prepare for the target performance objectives ahead of operation in ITER. The AGPS is a special switching power supply with demanding requirements: high rated power (55 MW), extremely high output voltage (-1MV dc), long duration pulses up to 1 hour and a unconventional operational scenario, where frequent short-circuits of the acceleration grids must be withheld by the system. The procurement of the AGPS is split in two: the low voltage Conversion System (AGPS-CS,) procured by the European domestic agency, and the high voltage DC Generators (AGPS-DCG), procured by the Japanese domestic agency. This paper deals with the AGPS-CS. Being an unconventional system, in-depth studies were carried out in the past years to assess the feasibility of the requirements, produce the functional specifications for the procurement and work out a suitable reference design. The executive design phase by Nidec-ASI, who was awarded the contract late 2015, is presently in progress; the basic choices of the reference scheme, composed of an input ac/dc converter connected via capacitive dc link to five Neutral Point Clamped (NPC) inverters, were endorsed by the Manufacturer. All the design details are now being finalized and suitable models developed to verify the system performance by means of numerical simulations. Special attention is devoted to the inverter modules, based on 6.5kV Integrated Gate Commutated Thyristors, which must comply with severe conditions in case of internal faults, due to the large amount of energy stored in the DC link capacitors. The paper will present the analyses to finalize the AGPS-CS design and will discuss the main design choices, in particular concerning the internal fault protections.
Progress and further plans towards high power negative ion beams at ELISE

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The negative ion source test facility ELISE represents the first step in the European R&D roadmap for the neutral beam injection (NBI) systems of ITER in order to consolidate the design and to gain early experience with a large and modular Radio Frequency (RF) negative ion source. The aim of ELISE is to demonstrate the ITER requirements with respect to extracted negative hydrogen densities \((329 \ A/m^2 \ H)\) and \((286 \ A/m^2 \ D)\) at an electron-to-ion ratio below one, a source pressure of 0.3 Pa and a beam homogeneity within 10%. The plasma of ELISE is generated by four RF drivers which illuminate half the ITER area, \((1\times0.9 \ m^2)\) with an extraction area of \(0.1 \ m^2\) (640 apertures, 14 mm diameter). Restrictions of the power supply allow beam extraction and acceleration up to 60 kV for 10s every 3 min during a continuous plasma operation up to 1h. After three years of operation ELISE has shown remarkable progress: 1 hour plasma discharges with repetitive 10s beam blips every 3 min could be demonstrated with current densities of \(94 \ A/m^2 \) \((H)\) and \(57 \ A/m^2 \) \((D)\) and an electron-to-ion-ratio below 1 at a moderate RF power of 20 kW/driver. Further improvement of the source performance by increasing the RF power is limited by the amount and the temporal stability of co-extracted electrons especially in deuterium operation. Alternative magnetic filter field configurations with different field topology are under evaluation to improve this behaviour and show first promising results. Increasing the RF power is further limited by the thermal heating of RF components and RF break downs which occur randomly around the drivers even at low pressures \((10^{-6}-6 \ mbar)\). Additional cooling and alternative matching configurations are investigated presently to explore the source performance in the high RF power regime.
Preparation of the ELISE test facility for long-pulse extraction of negative ion beams

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The test facility ELISE (Extraction from a Large Ion Source Experiment) at IPP Garching, Germany, aims to demonstrate ITER-relevant negative ion beam parameters which are required for the NBI system of ITER. ELISE is equipped with a Radio Frequency driven source and an ITER-like extraction system with half the ITER size. An $\text{H}^-$- or $\text{D}^-$- beam can be extracted for 10 s every 3 minutes from the continuously operating plasma source. The duration of the beam pulses is currently limited by the power supplies available at IPP. Although up to now record-setting 1 hour plasmas have been produced in $\text{H}^-$ as well as in $\text{D}^-$, long plasma pulse operation with multiple beam blips showed a key issue: the co-extracted electron current during the extraction phase is strongly dynamic and temporally instable, particularly in $\text{D}^-$. These instabilities are likely caused by back-streaming ions and Cs dynamics in the source [1]. In order to investigate the source physics in long beam pulses, an upgrade of ELISE using a cw high voltage power supply is envisaged. This upgrade requires a new cw diagnostic calorimeter for which a few concepts are being investigated, which make use of several thermocouples, IR thermography and water calorimetry to measure beam intensity, divergence, profile and homogeneity. In addition the suitability of a tungsten wire calorimeter to characterize the cw beam is being examined. Shielding of delicate components in the beam line, e.g. a large DN 1250 mm gate valve, by means of suitable protection scrapers, is being considered. These technical solutions are presented and discussed in the paper. Keywords: ITER, NBI, Negative Ion Source, RF Source, Beam Calorimeter [1] Fantz et al, Rev. Sci. Instrum. 87, 02B307 (2016)
Development of prototype elements for beamline components for ITER DNB and Indian test facility

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The Beam Line Components (BLCs) for the ITER Diagnostic Neutral Beam (DNB) and Indian Test Facility (INTF) are mainly water cooled elements made from CuCrZr which are designed to absorb heat flux up to 10MW/m² (e.g. Heat Transfer Element for calorimeter) according to their position in beam line. The design of these components imposes stringent requirements of having the specific chemistry of base material. Further, manufacturing involves dissimilar material welding and specific design for welds to enable full volumetric examination of the weld joint, tight tolerances on deep drilling of coolant passage over a length of ~2m, etc. The manufacturing process sequencing needs to be optimized with respect to the process and the corresponding material conditions. To address the above requirements, full scale technology development program has been undertaken which includes: i) development of the CuCrZr material with the unique melting technology to have the desired embrittlement, chemical and mechanical properties along with defining the aging cycle. ii) deep drilling of cooling channels of dia. 16 mm within 18mm thick plate over a length of 1500 mm with maintaining a drift less than 0.5 mm to achieve the designed cooling efficiency iii) Electron Beam welding of various dis-similar material combination like CuCrZr- Ni, OF Copper –Ni & Ni-SS which can further be inspected 100% with volumetric examination techniques. In addition to this, practical manufacturing problems have been addressed in terms of overcoming the deformation during the thermo-mechanical treatments of elements by manufacturing process sequence optimization. The paper shall present the consolidated results of various experiments during this development which has been carried out at NFTDC, Hyderabad, India. This indigenous development is expected to be used as a guideline during the manufacturing of Heat Transfer Elements for Calorimeters, panels for Neutralizer and tube elements for RID.
Manufacturing technology development for an ‘Angled’ accelerator grid segment for Diagnostic Neutral Beam (DNB) source

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The acceleration system of Beam Source (BS) of Neutral Beam (NB) system is composed of water cooled Oxygen-Free Copper multi-aperture grid systems which is designed for focusing of beamlets to a focal point located at distance>20m from the Grounded Grid. For present application in the accelerator for DNB, this focusing is obtained using a combination of segment bending and aperture offsets. In vertical direction, segments(1&3) and(2&4) are bent by 0.549° and 1.647° respectively. In horizontal direction, segment is shaped in horizontal direction (over length of ~825mm) to have two stage angles (i.e.0.222°,0.665°) on each side of centerline and to be fixed on a frame with flat plane, which therefore forms a ‘six-planed’ grid. Manufacturing of such ‘Bend Segment’ with stringent functional demands of tolerances on positioning (50µm), flatness (40µm) and angle (+/-0.002°) has been undertaken for the first time to the best of authors’ knowledge. Further, geometrically complex aperture shape is to be machined perpendicular to its own plane with very thin material left (1mm) after milling of water cooling channels and scooping of material for current distribution (for Plasma Grid (PG)). Therefore, need arose to establish a manufacturing methodology along with the impact and interdependence of various operations (i.e. milling of water cooling channel, aperture drilling, copper electro-deposition, material scooping, angled machining to achieve desired angle, and intermediate stress relieving). To address above issues, a full scale prototype of PG has been manufactured and significant data is now available on manufacturing tolerances and handling of ‘angled grid’. The paper shall present the technical data generated out of manufacturing of this prototype, summarizing the recommendations for real grid production on: optimization of the sequence of manufacturing, effect of each of the operations, post-manufacturing handling and identifying the measurement techniques. The experience gathered here provides a recipe for the best manufacturing practices for the accelerator of NB system for ITER and upcoming devices.
Progress on design and manufacturing of DC ultra-high voltage component for ITER NBI

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Design and manufacturing of DC 1 MV components have progressed for the ITER neutral beam injector. A multi-conductor DC 1 MV transmission line (TL) which can transmit five different voltages of 200 kV step simultaneously has been manufactured and tested. The TL is a gas insulation tube with SF6 gas of 0.6 MPa. A layout of those conductors inside the tube was designed through electric field analysis to suppress electric field concentration lower than 20 kV/mm. A high voltage insulation test of the TL at DC 1200 kV for 1 hour has been successfully performed. Cooling water supply system with insulation of DC 1 MV is also developed. A pure hot water feeding channel of 180 °C to the 1 MV potential is required to enhance the negative ion production in the ion source. A resistivity of the pure water decreases with an increase of the temperature. Low resistivity around 180 °C could result in a high leak current which causes further joule heating, however, the resistivity of hot water over 100 °C was unknown and the system cannot be designed. Thus, the resistivity of pure water up to 180 °C was experimentally investigated. As a result, high-temperature water channel with 10 mm diameter of insulation tubes was designed where a temperature rise is confirmed as small as 7 °C. All conductors and tubes are introduced into the vacuum through the high voltage (HV) bushing. Those are electrostatically shielded with five-layered coaxial electrode called as electrostatic screen. Through an experimental study on vacuum insulation, a scaling of voltage holding capability of multi-layered electrodes on the surface area was obtained. Based on the scaling, the HV bushing with five electrostatic screens was designed and manufactured, and voltage holding at 240 kV in each gap was successfully achieved.
Heating status and plasma performance in KSTAR

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The main mission of KSTAR program is exploring the physics and technologies of high performance steady state tokamak operation that are essential for future fusion reactor. Since the successful long pulse operation of 25sec at 0.5MA exceeding conventional tokamak capabilities in 2013, the duration of H-mode has been extended to over 50s which corresponds to a few times of current diffusion time. In addition to long-pulse operation, the plasma performance is further extended on the high betap discharge which characterizes the fully non-inductive discharge over 10s and achieves both ion and electron temperature over 5 keV simultaneously in line integrated density of 5*10¹⁹/m³. For supporting above KSTAR performances, main heating consists of various asset of heating mixture. Especially highly tangential 6 MW NBI consisting of three ion sources and 1 MW ECCD has been played in very important role in sustaining and achieving high performance plasma as well as the startup. Innovative concept of current drive, helicon, was installed and its preliminary results shows its efficient coupling and high power system will be tested in 2016. In addition to heating systems, various diagnostics including neutron is also used for characterizing and interpreting the plasma performance. A unique 2D/3D ECE imaging diagnostics on KSTAR provided the basic underling physics of the ELMs validated with the synthetic image based on the BOUT++ code and visualized TM/NTM mode phenomena during NTM control at high beta operation. Including above topics, the presentation will address the recent results on high ion temperature and neutron production discharge with the energy spectra of neutron and near term plan of heating upgrade in KSTAR.
Experimental results of helicon wave coupling using traveling wave antennas in KSTAR plasmas

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Helicon wave coupling for efficient off-axis current drive using a traveling wave antenna has been proposed. It was found that helicon wave can drive plasma current in the mid-radius of high electron beta plasmas in medium and large size tokamak due to moderate optical thickness and wave alignment nature of helicon wave in helical magnetic field. KSTAR tokamak can be a good platform to test this current drive concept because it has adequate machine parameters. In 2015 KSTAR experiments, mock-up traveling wave antenna (TWA) has been designed using electromagnetic 3D HFSS. Mock-up TWA was fabricated and tested by using a vector network analyzer (VNA). After TWA installation at the KSTAR in 2015, wave couplings were investigated by measuring and analyzing reflection and traveling loss at port 1 and 2 of TWA using VNA. Preliminary results of couplings between plasmas and TWA were obtained in both L- and H-mode plasmas (shot 14190 and 14364), respectively. The coupling of 70–80% could be controlled by changing radial outer gap, the distance between LCFS and poloidal limiter at outer mid-plane, without severe degradation of plasma confinement. In order to investigate helicon wave coupling using high RF power, prototype TWA and high power transmission line system are under fabrication. During 2016 plasmas experiments, RF power from 4 klystrons generating 75 kW at 500 MHz will be combined using three 6-1/8 inch coaxial hybrid couplers and transmitted through coaxial transmission line of 10 m long. 500 MHz, 250 kW RF power will be fed into prototype TWA connected to 5 inch coaxial feeders through dual alumina disk windows. Design and RF test results of key components such as prototype TWA, coaxial hybrid combiners and dual disk windows will be presented. Preliminary experimental results for high power helicon wave coupling in KSTAR plasmas will be discussed.
Steady-state operation of a DEMO-like tokamak requires substantial off-axis current be driven by external current drive systems. Non-inductive current drive is needed to complement the bootstrap current to support the plasma current in steady state. Recently, helicon wave current drive at frequencies of 500–700 MHz is gained much attention to achieve off-axis current drive with high efficiency. Helicon wave current drive has been shown computationally to have higher efficiency in the mid-radius region than other current drive techniques under reactor-like conditions. KSTAR can be a good test-stand to validate and to use off-axis current drive capability of helicon wave because it has adequate machine parameters and it will have high electron beta plasmas in near future with additional ECH power. To test the feasibility of helicon wave current drive, low power mock-up travelling wave antenna (TWA) has been fabricated and tested in KSTAR plasmas. The TWA has advantage, its insensitivity to the density profile at the plasma edge. Among the various TWA types, combline filter based TWA was chosen to test in KSTAR plasmas. It already had shown good load resilient and efficient coupling properties in JFT-2M tokamak. In order to investigate the effectiveness of off-axis current drive high power TWA with a launched $n||$ spectrum peaked at 3.0 will be installed and tested during 2016 KSTAR plasma experiments. The prototype TWA is composed of ten current straps, 8° tilted faraday shield, and 5 inches RF coaxial feeder for high power handling. The total length of the prototype TWA is 600 mm. The operation frequency of prototype antenna is 500 MHz and its bandwidth is about 40 MHz. Detailed design parameters and electrical properties of high power TWA for KSTAR helicon wave current drive will be presented.
Development of 4-MW KSTAR LHCD system

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The KSTAR LHCD system is to be upgraded for RF power up to 4 MW in 2020. The basic configuration of the system is composed of eight 5-GHz 500-kW CW klystrons, low-loss transmission line with oversized circular waveguide, and PAM launcher for the mid-plane injection. An off mid-plane injection near the upper diverter is also under consideration. A preliminary study based on a mid-plane PAM launcher will be presented in this paper. The transmission line will be oversized circular waveguides propagating in TE11 or TE01 modes. For TE11 mode, it is easy to design a mode converter, but the polarization rotation can be a problem. On the other hand, TE01 mode has circular electric field with no polarization and extremely low-loss in highly oversized waveguides, but the mode converter is complicated. Two types of TE01 mode converters were designed using HFSS. One is a serpentine type converting TE11circular-to-TE01circular mode. The diameter and total length of the mode conversion region is 80 and 462.8 mm, respectively. The conversion efficiency is 98.7\%. The other type has side wall injections of rectangular to circular waveguide and has high efficiency of 99.2\%. Mock-ups are under construction. Basic structure of the PAM is similar to that of Tore Supra. The RF power divided into six in poloidal by 3-dB hybrid-splitter followed by TE10-to-TE30 mode converter and then four in toroidal direction by multijunction. Dimension of the active and passive waveguide is 58 mm x 7 mm with 2-mm septum for $n/=2.5$. HFSS calculation showed that the designed multijunction has $S_{11}<-34$ dB, $S_{21} = -6.02\pm0.02$ dB, phase shift between adjacent port $= 90\pm0.8$ degree. Detailed RF design of 3-dB hybrid-splitter, TE10-to-TE30 mode converter and multijunction will be illustrated. Spectrum and basic characteristics for grill, FAM, and PAM type antenna were calculated and compared using the ALOHA code.
P2.033

Mode Converters for Low-loss Transmission-line of KSTAR LHCD system

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The KSTAR LHCD system is using a 5-GHz, 0.5-MW c. w. klystron and oversized rectangular waveguides. The WR187 output waveguide of the klystron transmits the RF power to the LH launcher via 80-m of transmission line composed of WR284 oversized rectangular waveguide. The overall transmission loss was about 34% including 26% of Ohmic loss. In order to transfer RF power effectively from a klystron to the launcher, low loss transmission-lines should be used. The resistive loss of the transmission line can be extremely reduced by adopting oversized circular waveguide. The Ohmic loss of TE01 circular mode decreased as the radius of the waveguide increases. In this case, mode converters are required to connect the rectangular output waveguide of klystron to the circular waveguides. Two types of mode converters were designed. One is the side wall type mode converter that converts the TE10 rectangular mode to TE01 circular mode. The other is the serpentine type mode converter which converts the TE11 circular mode to TE01 circular mode. TE11 circular mode can be easily excited by simple adiabatic transform of rectangular to circular waveguide. Mock-ups of two types of mode converters are under construction. In this paper, it will be presented that the comparison of the HFSS design and the measurement of the two types of mode converters. The transmission efficiency from TE10 rectangular to TE01 circular mode of side wall coupling and serpentine mode converters are -0.04 dB and -0.046 dB, respectively, according to the HFSS calculation. This mode converter design can be applied to the KSTAR LHCD system which is planned to be upgraded to 4-MW system with low loss transmission-line in 2020. * This research was supported by National R&D Program through NRF of Korea funded by MSIP (NRF-2014M1A7A1A02029891), BK21+ program and Korean ITER project.
A Metamateral Load For The LH Range Of Frequency In Magnetized Plasmas

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The coupling of lower hybrid (LH) range of frequencies waves to strongly magnetized plasmas is a critical issue on tokamaks as the RF power which can be transferred from the antenna to the plasma is often limited by the quality of this coupling. Development of new types of antennas aiming at improving the ability of the antenna to handle large powers in stationary conditions, as it will be requested on a fusion reactor, is hampered by the long and costly delay between the design and the feedback one gets from experiments on large facilities such as tokamaks. Moreover, the three-dimensional nature of the electron density in the vicinity of an LH antenna makes the analysis of the coupling difficult. Hyperbolic Metamaterials, based on piling up two-dimensional periodic structures, are characterized by a dielectric tensor which has a negative permittivity constant in one direction $\varepsilon_{//}$ and therefore can mimic an homogeneous plasma whose electron density $n$ is given by $n/nc=1-\varepsilon_{//}$, where $nc$ is the cut-off density of the wave ($1.7\times10^{19}$m$^{-3}$ at 3.7GHz). A fishnet load, composed of 26 grids layers, was designed with $\varepsilon_{//}=-3$, which corresponds to the optimal coupling conditions for a LH antenna. Each layer is a thin metallic grid (width 0.2mm, periods 28mm and 5.1mm) deposited on a 50mm-thick polyimide film. A low density foam ($\varepsilon=1.05$) is used as a spacer between the layers. The load was coupled to a 3.7GHz multi-junction type antenna module composed of 6 rectangular waveguides (8mm×72mm) aligned along the magnetic field. For an optimal distance of the load to the antenna, the power reflection coefficient measured in the waveguides (≈20% on average) and at the input of the module (≈1.5%) are consistent with the values computed with a full-wave code. Measurements of the electric field pattern in the load are also presented.
Advanced Control of Neutral Beam Injected Power in DIII-D


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In the DIII-D tokamak, one of the most powerful techniques to control the density, temperature and plasma rotation is by eight independently modulated neutral beam sources with a total power of 20 MW. The rapid modulation requires a high degree of reproducibility and precise control of the ion source plasma and beam acceleration voltage. Recent changes have been made to the controls to provide a new capability to smoothly vary the beam current and beam voltage during a discharge, while maintaining the modulation capability. The ion source plasma inside the arc chamber is controlled through feedback from the Langmuir probes measuring plasma density near the extraction end. To provide the new capability, the plasma control system (PCS) has been enabled to change the Langmuir probe set point and the beam voltage set point in real time. When the PCS varies the Langmuir set point, the plasma density is directly controlled in the arc chamber, thus changing the beam current (perveance) and power going into the tokamak. Alternately, the PCS can sweep the beam voltage set point by 10 kV or more and adjust the Langmuir probe setting to match, keeping the perveance constant and beam divergence at a minimum. This changes the beam power and average neutral particle energy, which changes deposition in the tokomak plasma. The ion separating magnetic field must accurately match the beam voltage to protect the beam line. To do this, the magnet current control accurately tracks the beam voltage set point. These new capabilities allow continuous in-shot variation of neutral beam ion energy to complement the discontinuous “on or off” modulation method presently used to control average beam power and torque input. This work is supported with General Atomics IR&D funding.
P2.036

Power and Particle Deposition Modeling of DIII-D and EAST Neutral Beam Systems

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The Neutral Beam system on DIII-D consists of eight ion sources. The basis of the DIII-D NB system is the Common Long Pulse Source (CLPS). The CLPS is an 80 kV high perveance, deuterium positive ion based system delivering up to 2.5 MW per source. The ion source is a filament driven magnetic bucket design and the accelerator is a slot and rail tetrode design with vertical focusing achieved through tilted grids. A similar neutral beam system with four sources is installed on the EAST machine in Hefei, China. DIII-D is in the process of enhancing the NBI system in several ways: by developing in-shot variable voltage capability, increasing NBI power through increased beam current, and increasing maximum co-injected and off-axis injected power by reconfiguring one beamline for co- and counter-injection as well as off-axis injection. Also proposed is a novel system to modulate the beam at high frequencies (~100 kHz, ΔV ≤ 1 kV) so the beam effectively becomes an antenna localized to the plasma core. The EAST development efforts are concentrated on achieving long pulse steady state operation. In support of these upgrades, two beam codes have been developed as tools to determine the power loading and particle trajectories onto beamline components and these results are used to determine operating limits and identify risks. The codes are benchmarked with respect to calorimetry data with further validation of the model input data by Doppler shifted spectroscopy. Predictions of power loading on key components of the DIII-D and EAST beamlines are made. A speculative hypothesis is presented suggesting that anomalous power deposition in the magnet region is a result of space charge effects. This effect and its implications for long pulse operation are discussed. *This work supported in part by the US DOE under DE-FC02-04ER54698¹ and DE-AC02-09CH11466.
Advances in technology and high power performance of the ECH system on DIII-D

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The gyrotron complex on DIII-D has been updated and comprises six gyrotrons installed and routinely operating reliably for injection of up to 3.6 MW into the plasma. The operational maximum of 5 s pulse length for the six gyrotrons allows up to 18 MJ total energy to be injected into the plasma. Recent system upgrades include faster launcher mirror scans and control by the plasma control system, a new 117.5 GHz operating frequency, and new 4-port power monitors. The eight sets of real time steerable mirrors that are installed on DIII-D inject the rf power from the tokamak low field side. The mirrors control the ECH deposition location. The real time steering feature with feedback is now routinely used for neoclassical tearing mode (NTM) experiments. The launched rf can be directed over ±20° from perpendicular toroidally, or steered poloidally over 40° in 200 ms. The transmission efficiency in the six operating 31.75 mm diameter corrugated waveguide lines is better than -1.1 dB. The measured HE1,1 mode content is over 85% for all the lines. Two grooved mirrors installed in miter bends are used for launching the arbitrary elliptical polarization required for different plasma configurations. The transmission line was upgraded with new rf power monitors at the last miter bend before the tokamak, which can measure wave polarization and mode content. Rf power reflected from the plasma is also monitored, and, together with visible light detectors, can give indication of low absorption in the plasma. A new design depressed collector gyrotron in the 1.5 MW class, operating at 117.5 GHz, is expected to be installed during 2016. Plans call for an expansion of the ECH system to ten gyrotrons including the new 1.5 MW gyrotrons operating at 117.5 GHz, providing up to 9.5 MW injected power.
The RAPTOR - RApid Transport simulatOR code [F. Felici et al 2011 Nucl. Fusion 51 083052] is a model-based control-oriented code that predicts Tokamak plasma profile evolution in real-time. One of its key applications is in a state observer, where the real-time predictions are combined with the measurements of the available diagnostics, yielding a complete estimate of the plasma profiles. The state observer RAPTOR is currently installed in the real-time control system of TCV, where it has been originally developed, ASDEX-Upgrade and recently RFX-mod. The last has pioneered its integration in the real-time MARTe - Multi-threaded Application Real-Time executor framework [G. Manduchi et al 2014 Fusion Eng. Des. 89 224], which will be the topic of this work. Thanks to this, RFX-mod can now contribute to develop integrated control techniques based on the state observer RAPTOR to avoid disruptions, which are highly reproducible in q(a)<2 RFX-mod Tokamak plasmas. Furthermore, coupling this code to the Tokamak equilibrium reconstruction code LIUQE [J. M. Moret et al 2015 Fusion Eng. Des. 91 1] is also currently under development at RFX-mod. Recently, RAPTOR has been applied to model ITER plasmas [J. Citrin et al Nucl. Fusion 2015 55 092001; J. van Dongen et al 2014 Plasma Phys. Control. Fusion 56 125008; E. Maljaars et al 2015 Nucl. Fusion 55 023001] and it is going to be extensively used to investigate control in DEMO, thus its real-time integration in present fusion devices, including RFX-mod, will promote its development and validation in view of the operation of the future fusion reactors. Acknowledgment. This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.
3D magnetic surfaces reconstruction in RFX-mod

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The Reversed Field Pinch configurations are characterized by strong asymmetries [1]; in order to prevent or mitigate possible consequent instabilities, suitable control systems are required. In RFX-mod (Padua, Italy), such a system includes a number of 192 saddle coils, independently controlled, fully covering the toroidal surface and operating in a coordinate strategy. An equal number of saddle probes provides the control system with the signals needed to close the feedback loop. Of course, the effectiveness of the control action depends on the capability to identify the actual 3D behaviour of the plasma column and to detect possible plasma-wall interactions [2]. Therefore, the classical identification procedures based on the axisymmetric assumption can be unsatisfactory and new methodologies able to detect the main 3D plasma characteristics are required. The paper shows the results achieved in a purely electromagnetic analysis of the experimental RFX-mod shots. The methodology is based on the equivalent 3D representation of the plasma current by means of stream functions and an analytical description of 3D magnetic surfaces based on the 3D interpolation of magnetic field lines. An effective use of high-performance computing architectures [3] is able to strongly reduce the required computational burden. [1] Bolzonella, T., and D. Terranova. “Magnetic fluctuation spectra and non-linear MHD mode interaction in RFX.” Plasma physics and controlled fusion 44.12 (2002): 2569. [2] Zanca, P., et al. “Plasma wall interactions in RFX-mod with virtual magnetic boundary.” Journal of nuclear materials 363 (2007): 733-737. [3] Chiariello, A. G., et al. “Effectiveness in 3-D magnetic field evaluation of complex magnets.” IEEE Trans. Mag 51.3 (2015).
3D electromagnetic analysis of the MHD control system in RFX-mod Upgrade

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RFX-mod is equipped with an advanced active control system of MHD instabilities, which consists of 48x4 saddle coils, housed inside a stainless steel Toroidal Support Structure, and 48x4 radial field sensor loops processed in real time to drive the currents in the control coils. Thanks to the high flexibility of this system [1], RFX-mod operations in the last years have allowed to reach the design plasma current of 2 MA in the RFP configuration and to investigate the very low q Tokamak regimes. In order to further extend the operational space of RFX-mod, a major upgrade of its magnetic front-end is now being studied. By removing the Inconel vacuum vessel, presently surrounding the plasma, the thin (3 mm) stabilizing copper shell will become the conductive surface closest to the plasma, thus decreasing the shell proximity and reducing the deformation of the Last Closed Magnetic Surface. The aim of this paper is the accurate calculation of the 3D magnetic field structure produced by the MHD active coils in RFX-mod Upgrade, in the presence of the new complex (geometrically and topologically) conducting structures surrounding the plasma, with particular emphasis on the evaluation of the toroidal coupling effects and the aliasing affecting the measured magnetic field harmonics. A 3D model has been developed, which includes the finest geometrical details of the conductive structures and two sets of non-axisymmetric field sources (48x4 saddle coils for MHD instabilities control and 2x11 saddle coils for local (poloidal gaps) error field control). A state of the art integral formulation [2] [3] is used to solve the eddy current problem in the frequency domain. [1] P. Zanca, et al., 2012 PPCF 54 (12) 124018–124027 [2] P. Bettini et al., 2015 IEEE TMAG 51 (3) 7203904 [3] P. Alotto et al., 2015 IEEE TMAG 10.1109/TMAG.2015.2488699
Modeling and mitigation of the magnetic field errors in RFX-mod Upgrade

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RFX [1] was originally designed with a load assembly consisting of a vacuum vessel (VV) and a thick aluminum stabilizing shell, with two poloidal and two equatorial cuts (i.e., gaps). After several years of experimental campaigns, a major modification of the RFX load assembly has been introduced [2], consisting in the substitution of the aluminum shell with a thin Copper Shell (CS) and the installation of a stainless steel Toroidal Supporting Structure (TSS). At the same time, the machine has been equipped with an innovative active control system of MHD modes [3]. After a relatively long period of successful operations, further modifications of the magnetic front-end of RFX-mod have been recently considered. In particular, the VV will be removed with the aim of improving passive MHD control (plasma-shell proximity increase) and plasma rotation (preventing the braking torque caused by the VV itself). In this paper, a detailed analysis of the new magnetic front-end is presented, with particular emphasis on modeling of the magnetic field errors generated at the poloidal gaps (butt-joint configuration) during the transient phases of the discharge. A non-linear equilibrium code, MAXFEA [4], has been used to simulate a reference scenario and provide the time evolution of PF coils and plasma currents as input of the 3D FEM analyses. A set of gap correction coils, previously installed in RFX [5], have also been included in the model to assess its capacity to minimize the error field at plasma edge, complementing the work presented in [6].

Development of Parallel Plasma Equilibrium Reconstruction in MAST

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The Parallel plasma equilibrium reconstruction code PEFIT [1], first developed for real-time plasma shape control of the EAST tokamak (and capable of one full equilibrium reconstruction in 300ms with a calculation grid size in 65x65) is being adapted for use on MAST. PEFIT is based upon the EFIT equilibrium code algorithm, but rewritten in C using the CUDA™ architecture in order to take advantage of massively parallel Graphical Processing Unit (GPU) processors to significantly accelerate the computation. This brings within reach the possibility of “full” equilibrium reconstruction within the real-time plasma control loop. Successful control using PEFIT/ISOFLUX has already been established in dedicated experiments on EAST. In addition to the real-time version, PEFIT provides detailed inter-shot analysis with a rapid turn-around. A PEFIT MDSplus module can be used to access diagnostic data and store the calculated results, or, alternatively, PEFIT is now interfaced to IDAM in order to be compatible with the MAST infrastructure and data model. A new flexible code customization technique (based upon XML) for different devices has been implemented into the code, making the development of PEFIT for MAST and other tokamaks much easier. On MAST, the total-induced passive current significantly affects sensor data and can adversely affect the reconstruction of the plasma equilibrium if not properly accounted for. A new induced current module based upon the lumped parameter circuit equation has been developed to estimate the induced currents within PEFIT. Preliminary testing indicates that results from PEFIT and EFIT++ are consistent with each other, with a single time-slice PEFIT reconstruction time of around 2ms for grid size in 65x65. Work is ongoing to optimize the algorithm further in order to satisfy the requirements of real-time plasma shape control. [1] X.N. Yue, et al, Plasma Phys. Control. Fusion 55, 085016, 2013
Development of divertor scenario for heat mitigation method based on charge separation in VEST

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Mitigation of heat and particle fluxes reaching on divertor plates is still a critical problem even though innovative divertor concept such as super-X and snowflake divertors have been suggested. A new divertor concept for the reduction of heat and particle fluxes is to convert thermal energy to electrical energy by separating electrons from the plasma with appropriate magnetic field. Feasibility study of charge separation has been conducted successfully in curved magnetic field [K.S. Chung et al., FED 2015]. However, this study has not been implemented in tokamak environment. In order to study feasibility of the charge separation in the VEST (Versatile Experiment Spherical Torus) device, a new scenario with two X-points is developed by using two partial solenoid (PF2) coils near the central solenoid (PF1). The location and geometry of divertor plates appropriate for studying a heat mitigation method based on charge separation are also considered. The details of diverted plasma scenario and future plan for the divertor heat mitigation experiments are presented in this paper.
Plasma internal profile control using IDA-PBC: Application to TCV

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The control of the safety factor q and/or the electronic temperature Te profiles is a key issue to achieve advanced plasma scenarios with high repeatability. This paper will discuss the new results of such plasma internal profile control on TCV, using total plasma current Ip, and ECCD heating source. The issue is that only the ECCD heating power is controlled, since the distributed heating profile has a pre-defined (Gaussian) shape. The control model is governed by the resistive diffusion equation coupled with a thermal transport equation. It is written in PCH (Port-Controlled Hamiltonian) formulation, where the system interconnection structure, dissipation and energy density are all explicit. Based on this model and from some reference values at some positions of q and Te profiles, the controller generates the admissible plasma profiles accompanying with a non-linear feedforward control. The feedback control based on a linear IDA-PBC (Interconnection and Damping Assignment - Passivity based Control) handles the convergence speed and the robustness by modifying the dissipation properties of the closed-loop system. The developed controller has been implemented into RAPTOR code for simulation tests before its integration in the TCV real-time control system. Two test scenarios are considered. The first one is based only on q-profile control, the total ECCD heating power is kept constant and only the difference between the ECCD sources which generates co-current and counter-current is controlled. While the second one extends to total plasma thermal energy control with the total ECCD power playing the role of control signal. Different profiles of ECCD for co-current and counter-current are taken into account and a PCH integrator is also added to successfully track the references in several test scenarios. The obtained results let us trust of a routine use of such plasma advanced control algorithm in the future for physic studies.
Architecture and platform of plasma control system T-15

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Presented work is related to the development and creation of hardware and software of Plasma Control System (PCS) platform of the modernized now tokamak T-15 [1] for the integration, configuration, testing and start-up algorithms for the calculation of electrical installation parameters, as well as for the modeling of the experiment scenario with taking into account of the real-time magnetic plasma control. The Mathworks Simulink is used as a main tool for modeling and synthesis of controllers together with National Instruments hardware and LabView RT software. We consider a multivariable control system that uses a multi-stage model design, which includes the construction of plant “Model”, the synthesis of “Controller”, the modeling of plant and control systems, the implementation of control system at the facility. Designed architecture of PCS T-15 operates in the simulation mode SIL (Software in the Loop) and control mode for different case of regulation. The proposed architecture allows to check and adjust PCS before the discharge on a tokamak, which increases the efficiency of the experiments while reducing costs.

Improvements of magnetic measurements for plasma control in KSTAR tokamak

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Noise width ($\delta V/V$) and drift level ($\Delta V/\Delta t$) in the magnetic measurements by using sensors such as magnetic field probes (MPs) and flux loops (FLs) has been fully satisfied with the requirements ($\delta V/V < 2\%$ and $(\Delta V/V)/\Delta t < 2\%$ for 60 s), for the plasma control in the KSTAR tokamak before the in-vessel control coil (IVCC) is used to control plasma shapes. From the experimental campaign of 2010, the in-vessel control coils (IVCC) is required for elongated plasmas (kappa > 1.7) to achieve the H-mode discharges in the KSTAR. However, there is large noise pick-up of few kHz in the magnetic measurements during the operation of the IVCC. Thus experimental investigations of the noise characteristics are carried out for the reduction of the noise. It is found that the noise comes from the IVCC power supply in the investigation of the noise characteristics. The noise width can be reduced up to less than 1 % of the signal amplitude by adding electronic filters between the electronic integrator and the data acquisition system in the signal path for the magnetic measurement. The plasma control during H-mode discharges has been improved by reducing the noise. In this work, the characteristic of noise due to the IVCC and the low pass filter system for reducing the noise will be described. Up to now, the improvements that have been achieved in the magnetic measurements by using the filter system for achieving better plasma control in the experimental campaign of 2016 in the KSTAR will be also presented. *This work was supported by the Korea Ministry of Science, ICT and Future Planning under the KSTAR project contract.
P2.047

Conceptual study of fast-swept divertor strike points suppressing ELM heat flux

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In order to avoid surface melting of divertor targets of big tokamak fusion reactors by localized ELM heat loads, we study a technique of spreading the flux by harmonic divertor strike point sweeping with a dedicated in-vessel twin-coil. If the sweep frequency gets above $1/t_{\text{ELM}}$ decay $\sim 300 \text{ Hz}$, local ELM plasma heat flux suppresses significantly (by factor $= 1 + 2 \lambda_{\text{sweep}} / \lambda_{\text{divetor}}$) where prediction for $\lambda_{\text{divetor}}$ ITER = 1-2 cm [1] is very thin (assuming midplane/divertor flux expansion of 10). Such a high frequency is the principal difference from the similar concept [2]. We ran dedicated Fiesta simulations for strike point harmonic sweep amplitude $\lambda_{\text{sweep}} = 7 \text{ cm}$, which requires coil current $I_{\text{sweep}} = 55 \text{ kA}$ for a device of the size and geometry of DEMO tokamak ($I_p = 21 \text{ MA}$, $B_0 = 6 \text{ T}$, $R_0 = 9 \text{ m}$). This could be achieved with 5 cm thick Aluminum coil twin (requiring water cooling of 2 MW ohmic losses), driven by a power source (14 kV, 1 kA) in a resonant circuit of the coil with 2.3 mF capacitor banks. The technique seems orders of magnitude less demanding on the coil current supply than alternative divertor concepts like the Snowflake, or Super-X divertor [3]. This 55 kA sweep coil would yield ELM heat flux suppression by factor $\sim 10$, which can be further enhanced by additional techniques (e.g. pellet injection, RMP ...). We discuss plans for experimental test on tokamak COMPASS (6 kAturns in 2 mm thick coil, driven by existing 1 kV, 1 kHz switching power supply), should yield factor $\sim 4$. [1] T. Eich et al. Phys. Rev. Letters 107, 215001 (2011) [2] M. Li et al. Fusion Engineering and Design 102 (2016) 50–58 [3] H. Reimerdes et al. 2015 EPS Conference on Plasma Physics, P4.117
Neutronics analysis for the ITER tritium and deposit monitor diagnostics

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This paper presents new results of neutronics analysis performed in support for the design development of the Tritium and Deposit Monitor (TDM) to be installed inside the ITER Equatorial Port Plug (EPP) #17. This monitor is a laser based diagnostics to provide information about the tritium content in the deposited layer on the inner baffle of the ITER divertor. Neutronics analysis is performed with the local MCNP models of EPP#17 which comprises two adjacent diagnostics: TDM and Core-Imaging X-ray Spectrometer (CIXS). The MCNP 3D models were converted from the corresponding CAD models. Critical neutronics issues related to radiation streaming, nuclear heating and activation are discussed and found shielding design solutions are presented in this paper. It was proposed to increase the length of one segment between two particular mirrors in the labyrinth optical mirror system to prevent direct neutron streaming from the plasma to the optical box in Port Interspace (PI), where maintenance is planned ~12 days after shutdown. Shield block made of boron carbide behind the optical box was added. To provide the possibility of personnel access to the PI area, Shut-Down Dose Rate (SDDR) has been calculated with the results presented as map distributions and estimates in spherical detectors. Using the Rigorous 2-Step mesh-based (R2Smesh) method of SDDR calculations which combines the FISPACT activation and MCNP transport allowed us to distinguish different decay gamma sources in forming the SDDR. The SDDR results are presented for two variants of the EPP17-CIXS models, with and without the monitor. Therefore the contribution to SDDR from the monitor was deduced. This work provides only the relative values of SDDR. The absolute values will be obtained after the finalizing of the designing work for all the EPP17 diagnostics. This is a forthcoming task of EPP17 diagnostics integration into the ITER-global C-lite MCNP model.
P2.049

Neutronics analysis of the in-vessel components of the ITER plasma-position reflectometry system

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The ITER Plasma Position Reflectometry (PPR) system will be used to estimate the distance between the position of the magnetic separatrix and the first-wall at four pre-defined locations, also known as gaps 3, 4, 5, and 6, complementing the magnetic diagnostics system. For gaps 4 and 6, the antennas are to be installed in-vessel between two blanket shield modules. The microwave signal is routed to/from the antennas using rectangular oversized waveguides that enter/exit the vacuum vessel through feed-outs located in upper ports 01 and 14, respectively. The antennas and adjacent waveguides are directly exposed to the plasma through cut-outs in the blanket shield modules and are therefore subject to high radiation doses from neutrons and photons, which may cause irradiation-induced changes in the material properties and compromise the integrity of these components. Here, we report on the preliminary neutronics analysis of these components of the PPR system using the Monte Carlo simulation code MCNP6, with the objective of estimating the thermal loads and amount of radiation damage. The first step consisted of translating the CAD models of the antennas and waveguides to MCNP6, using ANSYS SpaceClaim and the CAD-based modelling program MCAM. The resulting models were integrated in the ITER C-lite neutronics model provided by F4E and used to estimate the heat loads, DPAs and particle flux spectra in the components. The results, complemented with a finite element analysis carried out with ANSYS Mechanical, are presented in this paper. The operational temperature and the structural effects due to long-term irradiation are analyzed and discussed, and an evaluation is made on the necessity of prototyping some components to perform irradiation tests. The thermal loads obtained through the neutronics analysis presented here will be used as input in the global integrity analysis of the in-vessel components of the ITER PPR system.
Real-time software tools for the performance analysis of the ITER Radial Neutron Camera


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The Radial Neutron Camera (RNC) diagnostic is a neutron detection system with multiple collimators aiming at characterizing the neutron emission that will be produced by the ITER tokamak. The RNC plays a primary role for basic and advanced plasma control measurements and acts as backup for system machine protection measurements. To achieve its goals, the RNC diagnostic needs to acquire, process and store huge amounts of data per ITER discharge at high peak rates, calculating real-time measurements such as the neutron emissivity profile on the millisecond time scale. As a consequence, the data acquisition system and the diagnostic Fast Controller present several technical challenges and particular attention has to be given to the realtime firmware and software design. During the RNC system level design phase the following real-time data processing algorithms were developed to assess RNC data throughput needs and measurement performances: (i) real-time data compression block; (ii) real-time calculation of the neutron emissivity radial profile, based on Tikhonov regularization, starting from the line-integrated measurements, the line-of-sight geometry and using the magnetic flux information[1]; (iii) real-time calculation of the neutron emissivity profile using a-priori trained neural networks, the line-integrated measurements and the magnetic flux information (the best output from different neural networks being evaluated by a figure of merit that maps the neutron emissivity profile to the original line-integrated measurements)[2]. This paper presents results for the processing times of the various algorithms and their minimum control cycle for different conditions, such as number of lines of sight, number of magnetic flux surfaces and measurement error on the line integrated RNC measurements. References: [1] D.Marocco et al., Combined unfolding and spatial inversion of neutron camera measurements for ion temperature profile determination in ITER, Nuclear Fusion 51 (2011)053011. [2] M.Cecconello et al., Neural network implementation for ITER neutron emissivity profile recognition, this conference.
Nuclear Analysis of the ITER Radial Neutron Camera architectural options

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The ITER Radial Neutron Camera (RNC) is a multichannel detection system hosted in the Equatorial Port Plug 1 (EPP 1) designed to provide information on the neutron source total strength and emissivity profiles through the measurement of the uncollided neutron flux along a set of collimated lines of sight (LOS). Furthermore, the ion temperature profile and fuel ratio (nd/nt) can be assessed by means of line-integrated neutron spectral measurement. The RNC consists of two sub-systems based on a fan-shaped array of cylindrical collimators: the ex-port LOSs, covering the plasma core, embedded in a massive shielding block located in the Port Interspace, and the in-port LOSs distributed in two removable cassette integrated inside the Port Plug. Presently, the RNC layout development process is undergoing a System Level Design phase: several preliminary architectural options based on a System Engineering work have been defined for both the ex-port and in-port systems. A detailed nuclear analysis of these options has been performed through radiation transport calculations with the MCNP Monte Carlo code. The MCNP model of each RNC architectural option has been developed and recursively integrated in an upgraded version of the ITER MCNP C-lite model where all the details of the EPP 1 and nearby diagnostic systems have been included. Successively, the radiation environment at the detectors positions has been fully characterized through the evaluation of the expected neutron spectra and the secondary gamma background due to neutrons interactions with the surrounding structures. Moreover, the impact of a reduced ex-port shielding block on the neutron and gamma spectra has been investigated. The results of the present study provide guidelines for the development of the RNC final design and the necessary data for the measurement performance analysis.
The High Resolution Neutron Spectrometer (HRNS) system for ITER is an array of neutron spectrometers with the primary function to provide measurements of the fuel ion ratio, $n_T/n_D$, in the plasma core. Supplementary functions are to assist or provide information on fuel ion temperature and energy distributions of fuel ions and confined alpha-particles. The ITER requirement for the HRNS primary function is to obtain $n_T/n_D$ with 20% uncertainty and a time resolution of 100 ms. In this contribution, a conceptual HRNS system design and its measurement performance for $n_T/n_D$ will be presented. The HRNS system studied, is based on established instrumental techniques and its performance is assessed using realistic response functions for the individual spectrometers in the system. The main interfacing requirements for the HRNS is a 10 cm diameter aperture in the ITER first wall, tapered collimation resulting in a neutron flux in the order of $10^9$ n/cm$^2$/s on an area of 12 cm$^2$ at a distance of 16 m from the first wall. For optimum use of the available neutron flux, the system is divided into two sections: “low efficiency” neutron spectrometers in the front and “high efficiency” in the rear. Also investigated is to use an adjustable collimator between the front and rear spectrometers in order to enhance the dynamic range and overlap of the two sections. For the combination of neutron spectrometers presented here, it is shown that the system fulfills the ITER requirement, on $n_T/n_D$, over an order of magnitude in fusion power, $50 < P_{ fus} < 500$ MW. In the performance study, contributions due to neutron scattering in the vessel walls, collimator and beam dump are included together with a neutron induced gamma background.
2-mm microwave interferometer for COMPASS tokamak

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The COMPASS tokamak is equipped by the 2-mm microwave interferometer. This interferometer measures the electron density integrated along the central chord. Two VCO oscillators stabilized by the PLL together with multipliers generate two probing waves of the close frequency 139.3 and 140 GHz. The digital 2π-phase detector in the receiving part compares the phase between these probing waves. The resulting differential phase shift is 200 times smaller than the probing waves themselves. Therefore the phase response does not suffer from fringes for the full range of COMPASS electron density, which is more than 12x10^{19} m^{-3}. Both real-time and post-processed electron density take into account the interferometer line-of-sight, i.e. plasma shape, and correction to the non-linearity of the refractive index of the plasma. This way, the line-averaged electron density is obtained. The real-time signal is used for the electron density feedback.
Development and proof of concept measurements of the atomic beam probe diagnostic on COMPASS

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Atomic beam probe (ABP) is a diagnostic tool using a detection of ions coming from an ionized part of a diagnostic beam in tokamaks. The method allows measurements of plasma density fluctuations and fast variations in the poloidal magnetic field. Therefore, it gives the possibility to follow fast changes of edge plasma current, e.g., during ELMs in H-mode. The test detector has been installed on the COMPASS tokamak as an extension of the existing lithium beam diagnostic system. It uses a relatively simple set-up based on an array of conductive detection plates measuring the incident ion current, which is then amplified and converted to a voltage signal. The proof-of-concept measurements with the test detector has been done partly in laboratory and partly on the COMPASS tokamak. The concept and results of these measurements are the main targets of this contribution. For the proper interpretation of measured data the ion trajectories in the magnetic field of the COMPASS tokamak must be calculated by solving numerically the equations of motion. The ABPions code has been developed in MATLAB, in order to calculate positions of ions sequentially after every time step of the numerical scheme. TAIGA code is developed on the CUDA graphical processor, therefore, it is a parallel code calculating positions of ions, in the detector’s plane only. Results of simulations from both codes will be also presented. Test measurements with ABP test detector installed on the COMPASS tokamak supported with simulation codes clearly proved that measured signals on detector plates are caused mainly by Li-ions stemming from the diagnostic beam.
Implementation of Rapid Imaging System on the COMPASS tokamak

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The COMPASS tokamak has been recently equipped with two new fast color cameras Photron FASTCAM Mini UX100 operating in visible light. A new node, including both software and hardware, was developed for these cameras to ensure automatic and reliable operation integrated to the control and data acquisition system of COMPASS. The node provides camera function control, parameter setting, data transfer from the camera to PC, demosaicing, encoding of a preview video, data saving to the COMPASS database and managing of a disk space. The FASTCAM Mini UX100 camera operates at a full frame resolution of 1280 x 1024 pixels up to 4 kfps and is capable of achieving frame rates as high as 800 kfps at a reduced frame resolution of 640x 8 pixels. The camera uses 12-bit CMOS sensor with 10 \( \mu \)m square pixels allowing minimum exposure time of 1\( \mu \)s (shutter speed). A standard Bayer mask is used for color imaging. Images are collected to 4 GB internal memory, which limits maximum recording time. Node’s software is divided into three parts. The first part is programmed in C++. It takes care of direct communication with cameras, their control and raw mosaic data collecting. Then, raw data are stored to HDF5 files. The second part, written in Java, provides shot sequence control, cameras setting transfer, raw and video data transfer to the COMPASS database and disk space management. The third part demosaics raw data, processes and encodes a video. We use free open-source software AviSynth with our in-house plug-in that reads HDF5 files and demosaics it. The video is encoded with the MPEG-4 H.264 codec. Node’s hardware consists of the cameras equipped with a wide-angle lens combined with relay lenses, network accessories and the computer, all optically insulated from other tokamak systems. We introduce all mentioned subsystems implemented on COMPASS.
A new fast infrared camera Telops FAST-IR 2K was purchased on the COMPASS tokamak recently. It is equipped with a MWIR (medium wavelength infrared, 3-5 μm) InSb detector and is possible to reach framerate of 1.917 kHz in a full frame acquisition mode (320x256 px.) and up to 90 kHz in a sub-windowed acquisition (64x4 px.). The camera allows e.g. automatic exposure control, providing autonomous real-time control of an optimal exposure time to reach requested detector’s well filling, and other advanced acquisition control and data processing capabilities. The camera will be part of a new fast divertor thermography system with exceptional spatial resolution (≈0.6-1.1 mm/px. on the target plane, 0.04-0.12 mm/px. mapped to the outer midplane) with a possibility of measurements of radial profiles on divertor with 320x4 px. with a time resolution better than 20 μs. First experimental measurements of heat fluxes to the IWL (inner wall limiter) of the COMPASS tokamak using the new camera are presented as well as observations of the inner divertor region with exceptional temporal resolution. A design of the foreseen optical system for observations of the divertor region is described together with a design of a special divertor graphite tile used for the IR thermography, that will allow in-situ surface emissivity calibration.
Upgrade of the Compass tokamak microwave reflectometry system with I/Q modulation and detection

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The microwave reflectometry system on COMPASS tokamak uses the frequency modulated continuous wave (FM-CW) in K and Ka bands. The fast swept synthesizer together with the simple homodyne detection provides the complex beat frequency spectrum for the density profile reconstruction. The homodyne detection scheme limits the other applications like the Doppler reflectometry, therefore the scheme is rebuilt to the heterodyne system. The suitable way for the fast sweeping source is the implementation of the single sideband modulator (SSBM). Lack of suitable SSBMs on the market leads us to using of an I/Q modulator, which is modulated by the quadrature-phase signals. This contribution refers in detail to this upgrade, achieved results are shown too. With respect to the system parameters and availability of components, aspects like the choice of the modulating frequency, beat frequency, I/Q detector and others are clarified.
P2.061

Automatic ELM detection using g-SPRT on different diagnostic signals from the COMPASS tokamak

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The physics of Edge Localized Modes (ELM) is one of the most studied scientific fields in fusion research. Automatic detection of ELMs in different diagnostic signals is an important initial step during massive experimental data analysis. This contribution contains the description of the generalized Sequential Probability Ratio Test (g-SPRT) method used for automatic ELM detection in different diagnostic signals collected on the COMPASS tokamak. After determination of H-mode periods based on D-alpha signal in a given shot, ELMs are automatically detected in different diagnostic time traces (e.g. magnetic signals, Li-BES signals, divertor probe signals and also in D-alpha signals). The onset time, the maximum location and value, moreover the duration of each detected ELM is determined. g-SPRT is based on the classical SPRT, but instead of probability density functions, uses empirical distribution functions of studied phenomenon in diagnostic signals. Therefore approximation of empirical distribution functions of different phenomena in diagnostic signals has also been worked out. Analyzed diagnostic signals came from different radial positions of the studied plasma volume, thus from arrival times of given ELM into different detectors, the radial velocity of ELM is also estimated. Comparison of results from g-SPRT method with two different automatic ELM detection method (threshold technique and correlation technique), in different shots, will be also presented.
ArchiveDB WEB API: a web service for the experiment data archive of Wendelstein 7-X

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WENDELSTEIN 7-X and its superconducting coil system is designed for research on steady-state operation of stellarators. This sets high requirements on the control and data acquisition (CoDaC) system, with the archive database as one of its main components. W7-X ArchiveDB [1] is the central storage system for all engineering and scientific data. It stores raw data as well as processed data and provides a single database for a wide range of components and data types: diagnostic data from high-sampling ADC and video camera systems, as well as continuous data streams from the machine monitoring (incl. permanently running systems, like vacuum systems, cryosupply, and coils). Therefore the data acquisition is not limited to the period of single discharges, but it operates in a 24/7 manner. The WEB API provides a unified access to all data from ArchiveDB by using web service technologies. The API (Application Programming Interface) is based on the REST architecture style [2], using HTTP as the application protocol: all archived data is represented as resources, which can be addressed via URL. The users can read data in standard formats, like the compact JSON format or PNG for images. In the same way, the WEB API also provides a simple way to import new data to ArchiveDB, e.g., results from data analysis or data from diagnostic systems, which are still under commissioning. The usage of standards makes the API easy to integrate in all programming languages: currently, users call the API from C, Java, Python, MATLAB, IDL, and LabVIEW. A HTML representation allows browsing through the complete content of the archive by using a normal web browser, incl. preview plots of the data. This contribution will present the concept, implementation, and usage of the W7-X ArchiveDB WEB API.
A structural integrity monitoring tool for Wendelstein 7-X

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Wendelstein 7-X (W7-X) has been finally commissioned in 2015 and is now in its first stage of operation. Due to the complex structural design and a limited life time of some components, each step of W7-X commissioning and operation is carefully monitored by a considerable amount of different sensors. Unlike the fast machine control or the fast experiment data acquisition, the machine instrumentation (MI) works on a time scale of seconds to minutes and is not part of the W7-X control system. All MI data are captured and stored within the W7-X experiment archive. However the provided data browsing and reporting tools, like the Databrowser, are only of limited use when it comes to the approval of the structural integrity by mechanical engineers, due to the large volume of sensors and acquired data. The paper presents the W7-X approach for the on-line MI monitoring and structural integrity evaluation, based on the W7-X data archive and the incorporated JSON-based Web-API. It is shown how a client program, written in the high-level language MATLAB, allows the top-level supervision of multiple MI sensors and sensor types, while giving an easy access to individual sensors and signals, if necessary. The derived and implemented algorithms are a basis for a future deployment in a service oriented W7-X infrastructure. Furthermore it is presented how the program can be deployed for tasks beyond the initial scope.
Visual diagnostics at Wendelstein 7-X Stellerator

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WENDELSTEIN 7-X (W7-X) is a superconducting helical advanced stellarator which is currently in operation phase 1.1 at the Max-Planck-Institut für Plasmaphysik in Greifswald. During this startup period five uncooled inboard poloidal limiter structures made from fine corn graphite protect the plasma vessel wall, since the divertor, heat shields and carbon tiles are not installed yet. At 10 ports immersion tubes carrying each a set of two magnetic-field hardened cameras detecting light in the visible spectral range (VIS) and one near infrared camera (NIR) to observe the unshielded divertor-mounting components and the limiter structure. The two VIS-cameras are equipped with interference filters registering plasma radiation in particular spectral ranges of Ha, CII or CIII light to extract spatially-resolved information on particle flux and carbon production. Certain VIS-cameras are operated without filter with ultra-wide field of view objectives as video surveillance. The NIR lenses are surrounded with a ring of optical fibres for in-vessel illumination for inspection purposes. A rotating shutter protects the glasses from deposition and erosion. Further 10 overview video cameras are observing at tangential ports the plasma vessel interior for spatial and temporal inspection in the visible spectral range. Highly sensitive CMOS cameras have been applied to detect the vacuum magnetic flux surfaces during the commissioning of the device. Additionally a fast camera is setup at one port to detect fast events and plasma fluctuations. The presentation will give an overview of the observation systems and show first results obtained during the first operation phase.
Design and manufacturing progress of IRVIS endoscopes prototypes for W7-X divertor temperature monitoring

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The Wendelstein 7-X fusion device at Max-Planck-Institut für Plasma Physik (IPP) in Greifswald produced its first hydrogen plasma on 3rd February 2016. This marks the start of scientific operation. Wendelstein 7-X is to investigate this configuration’s suitability for use in a power plant. In order to allow for an early integral test of the main systems needed for plasma operation (magnet system, vacuum, plasma heating, control and data acquisition, etc), the divertor units and most of the carbon tiles covering the wall protection elements will be installed after the first operational phase OP1.1. For the later operation phases, the convective plasma heat fluxes will be distributed over a much larger area provided by the divertor target plates. An important diagnostic for W7-X will be thermography systems monitoring the surface temperature of the divertor target plates by collecting and processing infrared (IR) and visible (VIS) light from the divertor region of the plasma. For this purpose the company Thales SESO has been assigned to design, build, test and install 2 first prototypes of IRVIS (InfraRed-VISible) endoscope systems for the divertor of the W7-X Stellarator. Thermography is part of the operational and protective divertor diagnostics and has to detect signals indicating anomalous operation scenarios. The design of the horizontal and vertical target plates and the baffles in the divertor should keep the local power load below 10 MW/m². The current design of IRVIS endoscope is composed of four major elements: In-Vessel optical system (Cassegrain telescope system), ex-vessel optics (including dichroitic beam splitters, re-imaging optics and detectors), a cooled vacuum housing and an in-vessel shutter including drive and calibration equipment. The system is designed to operate under heavy-duty conditions. Design, integration, tests and manufacturing progress will be explicitly described in this paper.
The C/O monitor for W7-X will be a spectrometer of special construction with high throughput and high time resolution, suitable for controlling concentration of main impurities in plasma. The spectrometer will be fixed at horizontal position and at wavelengths corresponding to Lyman a lines of H-like ions of oxygen (at 1.9 nm), nitrogen (at 2.5 nm), carbon (at 3.4 nm) and boron (at 4.9 nm). Its purpose is fast monitoring of the spectral lines intensities which are reflecting level of impurities associated with general wall condition (oxygen), plasma-wall interaction (carbon, boron) or vacuum system leakage (nitrogen). The system will inform about the accumulation of the impurities and/or transient events associated with plasma wall interaction. The spectrometer will be composed of four independent channels, with individual dispersive elements and separate detectors. It will be constructed according to Johann geometry with Rowland circles radii adjusted to wavelength ranges registered by the respective channels. As a dispersive element for oxygen channel the TiAP crystal was chosen. Because in the range of spectra corresponding to the remaining spectral lines, the reflectivities of crystals as well as of gratings are very low it was decided to apply multilayer mirrors as dispersive elements. Multilayer for registering nitrogen spectral line will consist 150 layers of W/Si, for carbon line 90 Cr/Sc layers and for boron 100 Cr/C layers. Because of the irregular shape of the plasma and simple function of the spectrometer it was decided that the curvature of the mirrors ought to be cylindrical (instead of spherical, enabling imaging of the plasma). As detectors commercially available CCD designed for high energy detection will be applied. As an alternative detector, a proportional counter of special construction – a multistrip gaseous chamber (MSGC) is considered. The proposed system will be mechanically divided into two subspectrometers.
A multi-purpose manipulator (MPM) system is attached at an outer cryostat vessel port in the equatorial plane to transport electrical probes and targets to the edge of the inner vessel. From this parking position where the tip of the probe coincides with the inner vessel wall a fully controlled movement into the edge plasma for all magnetic field configurations is feasible. The distributed control system (DCS) is based on Siemens Process Control System (PCS7), which is the recommended standard for machine and diagnostic control at W7-X. The main features for the PLC system at the MPM are the production and control of UHV conditions in the target exchange and intermediate chamber, the operation of the separate linear motion units for slow and fast motion, respectively, the temperature monitoring of the probe head and custom target interface, the application of biasing voltage and gas injection. The PCS7 controls all functions of MPM operation and communicates with the W7-X central PLC. Inside the torus hall the operational status is visualized on a touch panel, where also predefined sequences, e.g. probe exchange, can be started. The whole function control for remote operation with plasma can be given to a virtual PC in the central diagnostic room. The parameters for stroke depth, velocity profile, biasing and gas injection are freely selectable within margins. The actions even for several strokes are triggered from the central timing of W7-X. The paper describes the layout of the electrical system and the control structure for the MPM with Siemens S7 under the framework of PCS7.
Software development for the simultaneous control of ten intelligent overview video cameras at W7-X

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In the past few years a ten channel video diagnostics system was developed, built and installed for Wendestein 7-X stellarator (W7-X). The system is based on EDICAM (Event Detection and Intelligent Camera) CMOS cameras (400 fps @ 1.3 Mpixel). In the first W7-X experimental campaign (OP1.1) the video diagnostic system is not integrated into the central control and data acquisition system of W7-X, therefore the development of a user friendly interface became necessary which had to fulfill several requirements. The system has to manage the setup all cameras (each with four independent region of interests), the measurement cycle and the real time storage of the collected data to SSD discs and has to provide live video stream of ten EDICAM cameras in the W7-X control room. The full bandwidth stream of the cameras results in a data rate of approximately 1GB/s for a single camera, and a huge amount of data when W7-X will operate for up-to 30 minutes. These requirements involves the development of a complex software package able to fulfill all the above mentioned tasks. The software package is organized into two stand-alone pieces: VIDACS (Video Diagnostics Data Acquisition and Control Software) which controls all cameras and EDIDAQ (EDICAM Data Acquisition Software) controlling the individual EDICAMs. This paper will present the detailed design, implementation, testing and the first operation experiences of this software package.
P2.069

Conceptual design of multi-foils system for stellarator W7-X

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Measurements of soft X-ray radiation from plasmas is a standard diagnostic which is used in many different fusion devices. Analysis of X-ray emission delivers among others, information about the electron density and temperature as well as can deliver an information about the impurity content in the plasma. The paper describes design of the soft X-ray diagnostic, multi-foil system (MFS,) for the stellarator Wendelstein 7-X (W7-X) operated at IPP-Greifswald in Germany.

The proposed diagnostic is based on a well-known foil-absorption technique which is use for the estimation of the electron temperature. The proposed diagnostic will be composed up to eight detector arrays (for five Si detectors each) with Beryllium filters of different thickness. The proposed detectors have 150 nm dead layer, 380 \( \mu m \) active layer and a sensitive area of 4.6\( \times \)4.6 mm. They have an electron collection time of about 5ns, a junction capacitance of 10pF (at reverse voltage \( UR = 50V \)) and a dark current below 12 nA at room temperature. The main MFS vacuum chamber will be mounted on a suitable port (AEN20) of Wendelstein 7-X via a gate valve. The port is tilted to the horizontal position by an angle of 53.5°. At the chamber entrance windows there are eight pinholes, covered by a Beryllium foil of 10 \( \mu m \) thickness in order to protect the detectors from ECRH stray radiation. Additionally, inside the MFS vacuum chamber, eight Beryllium filters of different thickness will be positioned to register signals in eight energy ranges as defined by filters. The preamplifiers to convert the small detector currents to a voltage signal will be directly attached to the air-side of the feedthrough flange in close vicinity to the detectors. They will consist of four printed circuit boards connected to the analog-to-digital converter via multi-wires coaxial cables.
Commissioning and first operation of the pulse-height analysis diagnostic on Wendelstein 7-X stellarator

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The Wendelstein 7-X (W7-X) stellarator started its operation at the end of 2015. The first operation phase is conducted both with helium and hydrogen as working gas and has achieved first plasmas in the order of 500ms at the time this abstract has been written. The initial experiments have also been devoted to commissioning, tests and optimization of diagnostic systems. In this paper we report on the commissioning of the Pulse Height Analysis (PHA) diagnostic. The PHA measures radiation from the plasma core in the energy range of 0.25 - 20 keV in order to estimate the electron temperature and core plasma impurity content. The PHA design is optimized for measurements in high performance plasmas in future operation phases of Wendelstein 7-X. The commissioning phase allowed to test functionalities and to validate the operation of the PHA. By applying the appropriate filters, each of the 3 channels observe the radiation spectrum in selected energy ranges. The plasma radiation is recorded by Silicon Drift Detectors, which have been specifically adapted to PHA with regard to their energy resolution better than 180eV. The temporal resolution of the PHA system is about 100 ms. The paper presents commissioning of the PHA system conducted in the first experimental campaign of W7-X. Findings are suffering at this stage from low densities and temperatures (with regard to counting rates) but the commissioning and characterization both in helium and hydrogen plasmas has been obtained and will be presented. Spectra from W7-X He- and H-plasmas will be also compare to simulation results obtained with the ray-X code.
Implementation of the soft X-ray multi-camera tomography diagnostic in the Wendelstein 7-X stellarator

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The quasi-steady state high power plasma experiments at Wendelstein 7-X are expected to become pioneering research benchmarking the advanced stellarator concept. The results will bring comparisons to the huge amount of experimental findings in other stellarator and tokamak devices. After the successful start of hydrogen plasmas in February 2016, the set of plasma diagnostics will be extended during the shutdown phase (between OP1.1 and OP1.2, to start in 2017). The soft X-ray multi-camera tomography system (XMCTS), a key diagnostic for the detection of high-frequency instabilities and MHD-mode dynamics, is planned to be included in OP 1.2. Twenty pinhole cameras aligned on a poloidal circumference are installed inside the plasma vessel within a segmented stainless steel support structure. Each camera is fitted with a silicon diode array (AXUV) and a curved beryllium filter (transparent for photon energies >1keV). For design, engineering and manufacturing of the mechanic and electronic components of the XMCTS an extensive set of requirements had to be met, that are special for operation in such a long discharge plasma device. This includes for instance the handling of sputtering of wall material, which can be comparable to one month of operation in usual short-pulsed fusion plasma devices and the need for active cooling. Especially here the welding procedure of the pipes (water cooling, signal) in the complex 3D geometry is very sophisticated. During plasma operation inaccuracies of the diagnostic from misalignment can result due to thermal movement of in vessel components. Other issues as electronic pickup during ECRH heating or the influence of the variances of the thicknesses of the Be-filter on the results of tomography are also discussed. This paper provides a summary of technical issues solved, electronic demands fulfilled and results from tomographic reconstruction of synthetic emissivity distributions modulated in time and in space assuming high-frequency MHD-modes.
Concept for calibrating the Rogowski coil system of the Wendelstein 7-X stellarator

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Thirteen Rogowski coils have been installed in the vacuum vessel of the stellarator Wendelstein 7-X (W 7-X). They are designed to measure the equilibrium plasma currents as Pfirsch-Schlüter current and bootstrap current. The coils will be calibrated using a conductor positioned inside the plasma vessel with an alternating current passing through it. The response of the coils is measured and compared to calculations. To reduce effects caused by the (frequency-dependent) induced currents in the vessel wall, the calibration will be done at alternating current frequencies up to 1 kHz. This paper reports on a multi-purpose design that combines a rail system for the neutron calibration with the conductor for the calibration of the Rogowski coils. The rail system is needed to carry a neutron source for the calibration of the neutron counters. The actual rail system is designed in such a way that it can be additionally used for running the calibration current for the Rogowski coils. While this works well for the continuous Rogowski coils where only the total current needs to be known, the segmented Rogowski coils need more care since they are designed to measure current density distributions. It is foreseen to install an additional calibration unit with a dedicated multi-loop setup.
A Multi-Purpose Manipulator system for W7-X as user facility for plasma edge investigation

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The investigation of edge plasmas at W7-X requires a flexible tool for integration of a variety of different diagnostics as e. g. electrical probes, probing magnetic coils, material collection, or material exposition probes, and gas injection. A multi-purpose manipulator (MPM) system has been developed and attached to the W7-X vessel before the operational phase 1.1. The system was designed as user facility for many diagnostics, which can be mounted on a unique interface. The manipulator system, located in the equatorial plane, transports the inserted diagnostic probe to the edge of the inner vacuum vessel. From there the probe can be moved over a maximum distance of 350 mm to different positions inside the plasma with a maximum acceleration and deceleration of 30 m/s\(^2\). Acceleration, speed and stroke depth are individually adjustable and programmable by a PLC system within predetermined limits. The MPM system can be equipped with multifunctional probes and is prepared for cooling/heating of the probe head, gas injection, and a flexible setting of the electrical diagnostic. In the paper the MPM components, their functionalities and the interface to the probe head are described. The scope of physical parameters is presented for the development of diagnostic probes applicable in the MPM user facility. This also includes the operational parameters for the movements and the electrical environment for measurements. The operation of the MPM during the OP1.1 is demonstrated on the application of a multipurpose probe head, including electric, magnetic and thermal probes.
Spatial Distribution of Accumulated Neutron Emission in KSTAR

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Long-pulse D-D plasma operation in the annual KSTAR plasma campaign is performed and involved Ohmic heating and auxiliary heating such as a neutral beam injection (NBI) of high power with deuterium beams. The NBI heating power reached up to 6 MW at the moment. In addition, many energetic runaway electrons are also observed through hard-X ray (HXR) monitoring during the operation. Runaway electrons of high energies can be high enough to produce photonuclear reactions in the walls from photo-nuclear processes. The photonuclear reactions due to runaway electrons with energies up to ~20 MeV are observed in the 2010 KSTAR plasma campaign. The intent of this study is to investigate its spatial distribution in accumulated neutron production in KSTAR. To carry out the work, four natural nickel (Ni) samples are installed inside the KSTAR vacuum vessel. The spatial distribution of accumulated neutron emission in KSTAR is determined by using neutron rates obtained from the samples based on activation analyses. The resulting measured neutron rates have anisotropic spatial distribution inside the KSTAR vacuum vessel. It appears that neutron production in the vacuum vessel is not only due to D-D beam-plasma reactions, and that both the NBI and the high energy X-rays could be attributed to neutron production from KSTAR. There is need to estimate factors influencing to neutron production in KSTAR for further study. More details will be presented.

Zeff profile measurement system by using Thomson Polychromator

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To measure Zeff profile, most plasma machine equipped brehmsstrahlung measurement system like as filterscope diagnostic. In KSTAR, however, a new type brehmsstrahlung measurement system developed and tested at single point in KSTAR 10th campaign in last year.\cite{1} In 2016 KSTAR campaign, to Zeff profile measurement, we expand this concepts of brehmsstrahlung measurement system to multi points; two for core and two for edge. Thus four polychromators are modified to the new type polychromator that include bandpass filters for Thomson scattering and brehmsstrahlung signal. In case of detection system that have five APDs for Thomson and one PMT for Zeff. To calculate Zeff by using brehmsstrahlung signal, calibration of wavelength and light intensity of radiation are necessary. The calibration is performed in the vacuum vessel and laboratory. In the vacuum vessel, a tungsten bulb and spectrometer are used, and in the laboratory, monochromator with tungsten light are used for wavelength calibration. By using new type polychromator system, we can measure $T_e$, $n_e$ and Zeff profiles simultaneous and at the same position. Moreover we can save a vacuum window for diagnostic system. In this research, explain the configuration of new type polychromator system and calibration result. Reference \cite{1} J.H. Lee, S.H. Lee, S.H. Son, W.H. Ko, D.C. See, I. Yamada, K.H. Her, J.S. Jeon and M.G. Bog, JINST, 10, C12012 (2015)
Design of phase contrast imaging on the HL-2A tokamak


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Abstract In this article we present the design of a phase contrast imaging (PCI) system on the HL-2A tokamak. This diagnostic is developed to infer line integrated plasma density fluctuations by measuring the phase shift of an expanded CO2 laser beam passing through magnetically confined high temperature plasmas. This system is designed to diagnose plasma density fluctuations with the maximum wavenumber of 80.64 cm$^{-1}$. The designed wavenumber resolution is 1.26 cm$^{-1}$, and the time resolution is higher than 0.2 $\mu$s. The broad ranging from 0.13 to 16.128 makes it suitable for turbulence measurement. Keywords PCI, phase plate, phase shift, density fluctuations.
Development of real-time display system to monitor plasma shape toward long-time discharge on QUEST

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In QUEST (Q-shu University Experiments with Steady-State Spherical Tokamak), the achievement of the steady-state operation for long time discharge is one of its project objectives. For the achievement of the long time discharge, the identification of the plasma shape and position in real-time is important during the operation of the tokamak. By observing the temporal behaviours of the plasma shape and position in real-time, the experimenters can get indications about what measures should be taken to achieve long time discharge on QUEST. In order to accomplish this goal, novel, efficient and precise real-time display system for plasma shaping and position monitor is going to develop on QUEST for the observation of the continuous behaviours of the plasma that is produced inside the vacuum vessel. For the remote participation to the experiments, the display system will have subsystem that can record the entire display frame of the display system as video files and recorded video files are accessible through online during the operation of the tokamak. In addition, the display system will have an emergency safety notification system to alarm the occurrences of any abnormal situations. For the construction of an effective real-time display system, precise and accurate data acquisitions from the magnetic measurements should be ensured for long time discharge. The magnetic signals of flux loops, magnetic probes and rogowski coils usually obtained as numerical integration of raw signals which are affected a critical drift error. This drift error should be removed from the acquired magnetic signals. The field-programmable gate array (FPGA) based reconfigurable real-time data acquisition module can be the solutions for acquiring magnetic signals. On the other hand, hall sensors may be the suitable candidate for accurate and precise data acquisitions from the magnetic measurements. Drift errors do not occur with hall sensors also in long-pulse operation.
P2.079

**Final design of the diagnostic calorimeter for the negative ion source SPIDER**

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This paper describes the final design of the Short-Time Retractable Instrumented Kalorimeter Experiment (STRIKE) for the SPIDER experiment (Source for Production of Ions of Deuterium Extracted from Radio frequency plasma) under construction at the Consorzio RFX premises. The STRIKE diagnostic will be used to characterise the SPIDER beam during short pulse operation (several seconds) to verify the degree of attainment of ITER requirements about the maximum allowed beam non-uniformity. After a preliminary design developed in the last few years, the complete STRIKE diagnostic system has been recently subjected to a final review and is now ready for construction. The main components of the system are: an array of 16 tiles made of Carbon Fibre Composite, which will be exposed to the high power density beam (of the order of 20 MW/m²) exiting from the SPIDER Beam Source; a supporting and positioning system, based on a set of structures made of stainless steel, with controlled moving systems, to be installed within the SPIDER vacuum vessel; a set of thermal, electric and thermo-graphic sensors to properly detect the operating conditions. The paper will focus in particular on the development of the engineering design of the supporting and positioning system, with a description of the relevant CAD and FEM analyses, and will give an overview of the complete system, with reference to the manufacture of CFC tile prototypes and to the choice of the complete set of sensors.
Experimental investigation of beam-target neutron emission at the ELISE neutral beam test facility

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Neutron measurements are proposed for the SPIDER/MITICA Neutral Beam Injection (NBI) prototypes in Padua. Neutron emission is here due to reactions between the beam and the adsorbed deuterons in the target and thus depends on the deuteron absorption level in the beam calorimeter. We have investigated such process at the “half size” ITER NBI ELISE facility of the Max-Planck Institut. A first measurement campaign was carried out in 2014 during the initial deuterium operations of ELISE with a liquid scintillator detector used as global neutron monitor. The collected data were generally in agreement with calculations based on the local mixing model of deuterium deposition in the copper target. However, deviations approaching 40% from the predicted neutron yield were also observed at the highest beam currents (10 A) and could be due to neglected effects such as spatial profile variations or physics mechanism beyond the local mixing model. In order to clarify the apparent discrepancy between experiment and predictions, a second high current (> 10 A) measurement campaign was held during deuterium operations in 2015 and took advantage of a significantly improved determination of the beam current and profile thanks to recent developments in the calorimetry and infra-red diagnostic systems. Our experimental results show that neutron emission is in general very sensitive to the Cesium conditioning phase of the radio-frequency source, which is an essential ingredient to obtain high currents with a negative ion source. In this contribution, we present a detailed analysis of the data focusing in particular on the relation between neutron emission and the beam current. The implications of our results for understanding neutron emission from beam-target reactions are finally addressed and used for an updated evaluation of the emission expected at SPIDER/MITICA. This work was set up in collaboration and financial support of F4E.
Design and Testing of Crowbar Protection System for the JT-60SA Superconducting Magnet Power Supplies

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JT-60SA is a Superconducting Tokamak in the framework of the Broader Approach Agreement between Europe and Japan. For this International Project, both the Italian National Agency for New Technology, Energy and Sustainable Economic Development (ENEA) and Commissariat à l’Énergie Atomique et aux Énergies alternatives (CEA) are providing ten AC/DC converters for the poloidal superconducting magnets (rated in the range \(\pm 20\, \text{kA}, \pm 1\, \text{kV}\)), two converters for fast plasma position control coils (rated \(\pm 5\, \text{kA}, \pm 1\, \text{kV}\)) and a converter for the toroidal superconducting magnets (rated \(+25.7\, \text{kA}, \pm 80\, \text{V}\)). Each converter is equipped with a crowbar protection system. These equipments are under construction and testing by Jema and Poseico industrial suppliers. The crowbar system has to carry the current flowing through superconducting coils, during faults or plasma disruptions, to protect the AC/DC converter and the superconducting coil from over-voltages even in case of earthquake events, it is considered as a safety relevant component and it is designed to be seismic resistant. The circuit configuration is based on a hybrid-making-switch, composed by a bidirectional-static-switch with a mechanical-switch and a varistor in parallel. The static-switch is composed by eight thyristors in bidirectional configuration (four connected in parallel and four in antiparallel), and it is designed to be able to safely intervene even if one thyristor or the mechanical-switch is not operating. The mechanical-switch (rated \(50\, \text{kA making-capacity}\)) takes some tens of milliseconds to close, limiting the conduction time of thyristors. The overvoltage protection is performed by a voltage transducer, a Break-Over-Diode and a varistor assuring a fast intervention with a high redundancy protection level. The current capability test of crowbar system required a maximum peak current of \(\pm 26\, \text{kA}\) and a maximum specific energy \(I^2t = 4.6\, \text{GA}^2\, \text{s}\) for TF SCMPS. Finally the crowbar system and DC bus-bars were subjected to seismic tests in laboratory.
Final tests of four switching network units procured by the European Union for JT-60SA

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Switching Network Units (SNUs) are inserted in the power supply circuits of modern tokamaks for plasma initiation. In the framework of the “Broader Approach” agreement, the four SNUs for the superconducting modules of the JT-60SA Central Solenoid will be procured by European Union through the Italian Agency ENEA. The design is based on the synchronized operations of a light electromechanical contactor and of a static switch to combine the benefits of both devices. The adopted solutions could be extended to many fusion and industrial applications. Exhaustive tests were performed in last years both on critical components and on the complete SNUs. In particular:

The first SNU was tested in the ENEA FTU facilities, even in conditions more demanding than expected during JT-60SA operations. The four SNUs, consisting of 24 cubicles, were assembled for operational tests in a configuration similar to the final installation and controlled by their Local Control Cubicle and by an emulator of the JT-60SA Supervising Computer.

A SNU was able to divert to a breakdown resistance (from 0.25 Ω to 3.75 Ω) currents up to 20 kA, producing a nominal breakdown voltage of 5 kV in less than 100 μs with a small overshooting voltage limited to about 200 V. Due to the obtained short duration of the arc in the contactor, the erosion is reduced with advantageous effect on the system lifecycle. A further electronic making switch allows a fast (less than 150 μs) resistance reduction to support the plasma ramp-up. The tests showed good performance repeatability for all the SNUs. The accuracy of all the breakdown resistors was well within ±2%. Specific tests showed that they could dissipate much more than 360 MJ. Since all the tests on the CS SNUs were successfully, the SNUs will be delivered to Japan in Spring 2016.
Effective control of Resistive-Wall-Mode (RWM) is mandatory in JT-60SA, the satellite tokamak under construction in Naka (Japan), since one of its main objectives is to reach steady-state high-beta plasmas. The RWM control system is based on a set of 18 in-vessel sector coils, placed on the plasma side of a conductive wall and individually fed by a dedicated fast power supply system (RWM-PS). For each coil, the RWM-PS has to produce arbitrary current waveforms following in real time the reference generated by the JT-60SA MHD Controller. The time response shall be fast enough to avoid excessive growing of plasma instability. Previous studies allowed establishing the main RWM-PS requirements, both in terms of output voltage and current ratings (240 V - 300 A) and in terms of dynamics. The latter in particular is very demanding: current bandwidth of 3 kHz in closed loop and latency between reference and output less than 50 ms. Possible technical solutions were explored in the past; the most convenient, based on 18 H-bridges was assumed as reference design once verified the feasibility of the fulfilment of the dynamic requirements. The development of a prototype was launched at this purpose and successful results were achieved thanks to the adoption of new hybrid Silicon-Silicon Carbide (Si-SiC) IGBT, driven by a fast full digital control board. The reference design approach with these switches was confirmed for the procurement of the whole system, presently in progress; the design phase is expected to be completed in summer 2016. This is the first PS system for fast control of plasma instabilities in fusion experiments adopting SiC semiconductors. The paper will give an overview of the whole final design with particular emphasis on the newest features and on the interface issues for an effective integration with the JT-60SA power and control systems.
Design of JT-60SA cryodistribution

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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. The JT-60SA helium refrigerator system (HRS) supplies supercritical or gaseous helium to cold components: superconducting coils, coil supporting structures, cryopumps, high temperature superconductor current leads (HTS CL), and thermal shields. The transfer line is a vacuum heat-insulation multiple piping and inserted to the tokamak hall from the cryogenic hall, where HRS is installed. Installation space around the tokamak is limited due to reuse of the building of JT-60U, heating instruments, diagnostics and maximization of plasma volume. Therefore, small 11 valve boxes are installed around the cryostat instead of a large distribution box. The valve box is cylindrical form: 2 m in height and 1.4 m in outer diameter. The transfer line connects directly the tokamak cryostat penetrating the wall of the tokamak hall. Cold helium in pipes from the transfer line distributes to cold components passing through in-cryostat piping, valve boxes, a cryopump valve unit, and coil terminal boxes, more than once. The HTS CL is placed in the coil terminal box and requires cold helium supply lines and a return line for 300 K. The 300 K line is returned to HRS apart from the transfer line. Some valve boxes have a safety valve unit. When a fast discharge occurs on superconducting coils, the safety valves are opened and the cold helium gas goes to a quench tank through a quench line. All lines mentioned above are required to satisfy criteria of the pressure drop and withstand the gravity load, the displacement due to vacuuming the cryostat and cooling down the cold components, and the seismic load. In this work, the design status and the manufacture progress of these cryodistribution lines are reported.
Completion of manufacturing of equilibrium field coils for the JT-60SA tokamak

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The programme of constructing JT-60SA device is progressing as a satellite tokamak of the Broader Approach project. JT-60SA has superconducting poloidal field (PF) coil system which is procured by JAEA, and 18 D-shaped toroidal field (TF) coils of which Europe has been in charged. PF coil system consists of a central solenoid (CS) with four solenoid modules and six circular coils which are utilized to produce the plasma equilibrium field (EF). Each of EF coil has individual diameters, 4.5 to 12 m. Fabrication of EF coil was started from EF4 which is set at the lowermost of torus in the beginning of 2009. EF5 and EF6 coils will be installed lower side of tokamak, so that these coils had to be preferentially manufactured before the setting of vacuum vessel was started. In the end of 2013, these 3 EF coils were completed and temporally set on the cryostat base. Now these coils are waiting for setting on the TF coils after these coils are fully installed. For the circular coils such as EF coils, circularity is very important issue to sustain the plasma with high performance. For EF4, EF5 and EF6 coils, the final circularities of winding pack were 1/6 to 1/10 smaller than the design tolerance that was calculated from the allowable error field to control plasmas. From summer 2014, EF1, EF2, and EF3 coils were started to be manufactured and will be completed by the end of summer 2016. Then, all the superconducting EF coils are finally manufactured. So far, the circularity of EF2 (9.2 m in diameter) is clear, 0.4 mm. This value is 1/15 of the design tolerance. It is expected that great accuracy of manufacturing will be successfully kept until completion of manufacturing.
In the framework of the Broader Approach program, ENEA is in charge of the in-kind supply of 18 Toroidal Field (TF) coil casings for the superconducting tokamak JT-60SA being assembled in Naka site, Japan. ENEA commissioned the company Walter Tosto (Chieti, Italy) the fabrication of two sets of 9 casings to be delivered to ASG Superconductors (Genoa, Italy) and GE (Belfort, France), in charge the following integration of the winding packs into the casings. The composition of the casing components and the detail design of the interfaces have been finalized under the coordination of Fusion for Energy (F4E) and the agreement of the other parties. Two different sets of mock-ups representative of the components have been realized to validate fabrication methods and special welding processes. The manufacturing activities have been divided into different production steps: composition of the components by cutting, forming and welding and then machining to the final shape. The first seven casings have been completed during 2015 and are being used for the completion of the TF coils. On the base of manufacturing experience of the first casing components and the completion of the first coils, the production process has been improved and the schedule optimized. This paper reports the status of manufacturing of the casings during 2016.
Thermohydraulic and Quench Behaviour of the JT-60SA Toroidal Field Coil in Cold Tests Facility

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The Toroidal Field system of the JT-60SA tokamak comprises 18 NbTi superconducting coils. In each TF coil (TFC), 6 Cable-In-Conduit Conductor (CICC) lengths are wound in 6 double-pancakes (DP) and carry a nominal current of 25.7 kA at a temperature of 5 K. These coils are tested in the Cold Test Facility (CTF, CEA Saclay), the test program including a quench for each of the first coils of the two series (ASG and GE). In order to ensure the tested magnet safety, a regular quench detection system is based on compensated voltages. A SuperMagnet (CryoSoft) model has been developed, each of the 12 Pancakes being modelled by THEA (Thermal, Hydraulic and Electric Analysis) with its own friction factor, heat exchange coefficient, magnetic field and heat input. The cryogenic circuit is modelled with the FLOWER code (Hydraulic Network Simulation) comprising the pump, heat exchanger, heater, relief valves, and quench tank. The experimental quench performed on C10 coil has been simulated, representing the increase of inlet helium temperature up to 7.46 K and the quench on side pancake DP6, followed by the safety current discharge. The calculation results are compared with the tests measurements signals: the helium temperature, pressure and mass flow at the extremities of the conductors and coil. The additional calculated parameters are presented: the conductor temperature, the current sharing temperature, the normal length, the resistance and the resistive voltage. Results regarding the external cryogenic loop are also detailed. Particular attention is paid to mass flow, pressure and temperature signals to assess the feasibility of a possible secondary quench detection of thermo-hydraulic nature in the JT-60SA Tokamak. Theses experiments and simulations can also help validating similar models used for ITER magnets quench studies and safe operation.
JT-60SA TF Coils procured by ENEA: an intermediate assessment

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ENEA, in the framework of Broader Approach program for the early realization of fusion with the construction of JT-60SA tokamak, has committed to procure 9 of the 18 TF coils of JT-60SA magnet system. Within 2016 six coils will be completed and delivered to the cold test facility in Saclay, France, for the final acceptance tests before their shipment to Naka site for the assembly. Manufacturing has been divided in two main production steps: winding pack (WP) manufacturing and integration into casing. All the nine WPs have been already completed on September 2015 and the final integration phase has started in 2015 for the first coil. The integration, in its turn, is composed of six sub-steps: insertion, welding, embedding impregnation, final machining of interface areas, He piping assembly and final acceptance tests. Each of these steps has been already accomplished and the first coils have been delivered to Saclay to undergo the cryogenic acceptance tests. This paper provides an overview and intermediate assessment of the contract that ENEA signed with ASG Superconductors for this supply. Main electrical, geometrical and fluidic results of the nine WP produced and of the TFCs completed so far are also reported.
The Toroidal Field system of the JT-60SA tokamak is composed of 18 NbTi superconducting coils. Half of them are provided by France within the Broader Approach Agreement. These coils are manufactured by General Electric (ex-Alstom) at Belfort, France. Each TF coil is composed of 6 cable-in-conduit conductor lengths, wound in double-pancakes, carrying a nominal current of 25.7 kA at a temperature of 4.5 K. These coils are being tested in the single coil configuration at the so-called Cold Test Facility (CEA/IRFU Saclay, France). The test program includes a progressive operating temperature increase at nominal current up to quench for the first coil of each series and a DC operation at nominal current for 1 hour slightly below the expected quench temperature for all coils. In addition, a fast safety discharge is triggered at nominal current and 4.5 K for all coils. The test of the first coil (C10) has shown that the coil casing was hard to cool-down and remained at about 20 K during the test, which puts in competition for quench location either the central pancakes (peak field) or the side-pancakes (warmer). The test procedure was modified so as to trigger the quench at peak field, but the quench location at 7.46 K helium inlet temperature was identified on a side double-pancake in C10. The paper presents the CEA analyses for the French coils tested in the first half of 2016 in light of the NbTi strand superconducting properties coming from the strand characterization program, of the unit lengths hydraulic tests carried out by F4E, and of the full-size conductor tests carried out by CEA in the SULTAN facility. These analyses involve cable thermal and electrical modelling developed at CEA and already used in ITER R&D, design and characterization programs.
Quench tests analysis of the first JT-60SA Toroidal Field coils

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In order to check the performance of the JT-60SA Toroidal Field (TF) coils and hence mitigate their possible fabrication risks, a series of tests have been carried out in the Cold Test Facility (CTF) at CEA Saclay in nominal conditions at 5 K and 25.7 kA. One major test performed is the so called “temperature margin test” during which the inlet helium temperature of the winding pack is increased slowly with automation for approaching the calculated current sharing temperature of 7.3K in the high field region. For the first coils, the temperature is increased till the coil quenches. When the coil begins to quench, a Fast Discharge (FD) is triggered by the Magnet Safety System (MSS) which has a detection threshold of 100 mV - 100 ms. This allows to release all the nominal current from the coil to a Dump Resistance of 6.2milliohms with a time constant of about 10s. In the same time, the evacuation valves are opened to avoid any damages by overpressure in the coil or in the circuits. During this test, the voltage of each Double Pancake (DP) is registered by at a frequency of 10 kHz. This paper will present a study of the thermo-hydraulic behavior of the JT-60 SA coils during a quench thanks to the data coming from several quench experiments. It will analyze in particular the quench propagation velocity, the dissipated power in the coil and in the helium, the heated volume, the pressure rise in the helium and the operation of the safety valves.
Manufacturing and acceptance by CEA of the first JT-60SA TF coils

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In the frame of the Broader Approach, CEA provides 9 + 1 spare TF coils for the JT-60SA tokamak. Mid 2011, a manufacturing contract was awarded to Alstom, Belfort, now General Electric. The first years were dedicated to the manufacturing process definition, the critical phases qualification through a set of 12 mockups, the manufacturing QA definition and the procurement and commissioning of the tooling. This done, the production started early 2014. The workshop is organized in 12 workstations. The production starts with the preparation of the conductors provided by F4E and the winding in a stack of 6 Double Pancakes (DPs) which constitutes the Winding Pack (WP). The joints to connect electrically the DPs and the impregnations finalize the WP. End 2014, acceptance tests related to geometry, tightness and electrical insulation validates the WP for the following integration operations. The casings made under contract to ENEA, are delivered as a set of components, including the U shaped straight and curved legs forming the D shape of the coil and the inner covers. After preparation, the WP is inserted vertically inside the casing straight leg lying horizontally. The curved leg and covers are then added to close the coil. The WP geometry and location in the casing are carefully controlled. The casing elements enclosing the WP are then welded either manually or by robots before performing the blocking impregnation. A final machining adjusts then the coil interfaces to the precise WP references. After insulated piping integration, the final coil acceptance in terms of geometry, tightness and insulation allowed the delivery of the first coil at the cold test facility end 2015. This paper describes the first TF coils manufacturing process, the status of the production and provides first insights into the manufacturing feedback and the first coils performance results.
Metrology of an additively manufactured coil frame for stellarators

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Coil casings and coil frames for stellarators are geometrically complex components at high accuracy. A method of additive manufacturing combined with fibre-reinforced resin casting has been recently experimented [1] for the fabrication of complex coil frames. The method is named 3Dformwork and consists of additive fabrication of a hollow thin shell which is filled with resins or other appropriate materials. However, the accuracy of additive manufacturing for coils frames is a concern. Accuracy of 0.1% or higher is required for components defining the magnetic field of the fusion device. In this framework, the deviation between drawing dimensions and real additively manufactured coil frames is investigated. A coil frame of around 0.35 m long was designed and fabricated by Selective Laser Sintering (SLS) additive manufacturing in polyamide. Subsequently, the part was measured by an accurate Coordinate Measuring Machine (CMM). A contraction of 0.4% in vertical dimension and 0.35% in one horizontal dimension was measured. Also, additive manufacturing of type Stereolithography (SLA), which is based on photopolymers, is being studied as a possible more accurate fabrication method. It may achieve accuracy around 0.1%. Thus, standard SLS additive manufacturing still achieves lower accuracy than required for stellarator coil frames and other high precision fusion components. The paper reports the features of the sample coil frames, the used metrology methodology, the performed measurements and the resulting geometrical deviations. [1] V. Queral, 3D-printed fusion components concepts and validation for the UST_2 stellarator, Fusion Engineering and Design 96–97 343–347, 2015.
The Quench-Detection-System of the fusion experiment Wendelstein 7-X detects quench events within the superconducting magnet system constructed of 50 non-planar and 20 planar coils, 14 current leads and the bus bars. In the event of a quench the QD-System triggers the power supply of the magnetic system to shut down. The QD-System monitors the superconducting system by 486 Quench-Detections-Units. The first step of the commissioning phase was to check the wiring between the QD-Units and the superconducting magnetic systems, current leads and bus bars. Not only the electrical connection had to be checked also the correct contacting points of all wires on the magnetic system had to be verified. The verification of the correct connecting points was important to prove the unbroken protection. The next commissioning step was to adjust the balance of the QD-Units voltage measuring bridge. Two layers of all planar and non-planar coils are monitored through this bridge. Because of design deviations an inductive difference exists between both layers. This asymmetry is compensated by balancing of this bridge. The superconducting sections of the current leads and bus bars are monitored by a simple voltage measuring. The last commissioning step was to parametrise the detection criteria. A quench is detected by a defined difference voltage level $U > 0$ and an integration time $T \geq 0$. If both criteria are fulfilled a quench event is detected and the magnetic power supply is triggered to shut down. The paper describes the background, boundary conditions, measurement method and results of the wire check. In detail is described the used proceeding to balance the QD-Units and the lessons learned as well as the identified and used quench detection parameters.
The magnet system of the stellarator fusion device Wendelstein 7-X (W7-X) is composed of three different groups of coil systems. The main magnetic field is created by a superconducting magnet system that is accompanied by two sets of normal conducting coil groups, the Control Coils inside the plasma vessel and the Trim Coils (TC) positioned outside of the cryostat. The TC system consists of five coils, power supplies, cooling systems, high current connections and superordinated control systems. There are four coils (type A) of equal shape; the fifth coil (type B) has a slightly different shape due to space restrictions at the assembly position. Five individual power supply units whose design is based on four-quadrant current converters using Insulated-Gate-Bipolar-Transistors power the TC with individual currents and ramp rates. This allows the correction of error fields and also increases the experimental flexibility by providing a means to balance the divertor heat loads among the five field periods. The TC with its instrumentation and the power supplies including the switchgear to be connected to the coils have been designed and built in collaboration between IPP, Germany and PPPL, USA, funded by the Department of Energy. The design and integration of all auxiliary systems like the grid station, cooling for coils and powers supplies, control system and the associated cabling and piping has been developed, manufactured and accomplished in 2014. This paper describes the results obtained during the integral commissioning of the TC system, the operational experiences during the first experimental campaign of W7-X and its impact on the physics program.
Numerical investigation on the steady temperature rise of a by-pass switch

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The quench protection switch (QPS) is very important for ensuring the safety of the PF and TF coils of a superconductive Tokomak. The main function of a QPS is to protect the magnet as the coil quench occurs. Besides, a QPS has to withstand almost all of the coil current of some tens of kA flowing through it for a long time in the normal operation condition. This task is undertaken by the by-pass switch (BPS), which is an indispensable component of a QPS, no matter what principle the QPS is based on. Therefore, the study of the temperature rise is very important for optimal design of a BPS. In this paper, a scheme of a high-current BPS is proposed. The BPS consists of eight pairs of contacts in parallel which have certain capability of withstanding arc erosion during current commutation out of the BPS. The steady temperature rise of the BPS is simulated in the condition of natural convection with the approach of computational fluid dynamics. Furthermore, the influence of the space between each individual contact is investigated. Simulation results indicate that there exists certain critical space, beyond which the mutual influence of adjacent contacts can be neglected. This critical space can be regarded as a reference in designing a BPS with the trade-off between the dimension of the BPS (and the QPS) and its current-carrying capacity. The scheme has been applied in an experimental prototype of BPS. Temperature-rise experiments have been conducted on the prototype. Basically, the simulated temperature rise agrees with the experimental results. The deviation between them is also discussed.
Numerical investigation on the current commutation process of a quench protection switch

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Superconducting magnet is one of the most crucial components in a superconducting Tokamak. During the normal operation stage, high current of some tens of kA flows through the magnet with large inductance of \( \sim 1 \text{H} \). Therefore, extremely large energy (\( \sim 0.1-10\text{GJ} \)) is stored in the magnet, which must be dissipated in the case of magnet quench in certain duration before the occurrence of local or even overall damage of the magnet. The task of a quench protection switch (QPS) is to transfer the high current from the low-resistance branch to another one with a dump resistance, through which the energy is dissipated. A QPS based on the principle of artificial current zero has been proposed. The full QPS consists of four branches, i.e., the by-pass switch (BPS), the main circuit breaker (MCB), the dump resistance (DR), and the commutation branch (CB), which includes a capacitor, an inductor, and a commutation switch connected in series. In present work, the detailed current commutation process after the current is transferred from the BPS to MCB is investigated by a circuit model. Then, the influence of the frequency of countercurrent and the value of the dump resistance on the characteristics of quench protection is analyzed. The simulation results indicate that the commutation duration can be decreased slightly by increasing the frequency of countercurrent. However, the decrease of frequency of countercurrent can lower the current decreasing rate before current zero of the MCB, which benefits a successful interruption of the MCB and the reliability of the QPS. Furthermore, it indicates that the voltage across the superconducting coil decreases with the decrease of the dump resistance. However, from another point of view, a larger dump resistance can reduce the overvoltage across the commutation switch, which also benefits a reliable protection.
Design optimization of Normal Heat Flux First Wall panels for ITER

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The Normal Heat Flux (NHF) First Wall (FW) panels consist of a series of fingers, which represent the elementary plasma facing units and are designed to withstand 15,000 cycles at 2 MW/m\textsuperscript{2}. The fingers are mechanically joined and supported by a back structural element called “supporting beam”. The structure of a finger is made of three different materials, stainless steel for the supporting structure, copper chromium zirconium for the heat sink and beryllium as armour material. Due to their location and to the interfaces with other systems (e.g. diagnostics, remote handling), the 215 NHF FW panels are distributed among 31 design variants. These variants can be divided in 13 main variants which are significantly different, and 18 minor variants which are small deviations from the corresponding main ones. The aim of this paper is to present recent work aiming to achieve a global optimization of NHF FW panels design, considering also simplifications being introduced following prototype manufacturing. With this objective, CAD detailed models are created in CATIA. Finite Element (FE) simulations are done, in order to confirm that the thermo-mechanical behaviour matches closely the baseline design that was validated during the final detailed design review in 2013. This paper describes the work performed to implement these design solutions in the CATIA 3D models and the results of their assessment obtained by means of ANSYS numerical simulations.
Beryllium bonding pre-qualification for the ITER First Wall procurement in EU Domestic Agency

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This paper describes the main activities carried out for the conclusion of the EU-DA prequalification process for the supply of Normal Heat Flux (NHF) First Wall (FW) panels to ITER. A key part of these activities is the High Heat Flux (HHF) testing of a reduced scale FW prototype (Semi-Prototype (SP)). This component is manufactured by the AREVA Company in France and has a dimension of 221 x 665 mm\textsuperscript{2}, corresponding to about 1/6 of a full-scale panel. Other EU manufacturers are also developing similar manufacturing routes in view of the future series production. The manufacturing process makes extensive use of Hot Isostatic Pressing joining method, which was developed over more than a decade during the ITER Engineering Design Activity phase. The SP includes some variants (tile grain orientation, anti-diffusion layer materials) which bring additional valuable knowledge beyond the qualification. The HHF test is performed successfully in the “Tsefey-M” facility at the Efremov Institute in Saint Petersburg, Russia. The objective of the testing is demonstrating successful bonding at up to 125% of the design heat load. The test protocol and facility qualification are presented and the thermal behavior under the 7500 cycles at 2 MW/m\textsuperscript{2}, and 1500 cycles at 2.5 MW/m\textsuperscript{2} is described in detail. A sufficient number of tiles succeeds in the testing (less than 20% temperature increase between initial and final heat cycles) for the two test levels of 2 and 2.5 MW/m\textsuperscript{2}, granting qualification as per ITER requirements. The successful qualification of the EUDA SP opens the way to full scale prototype qualification, which is currently underway. The full scale prototype program is the last milestone before series production of the 215 panels that EUDA is due to deliver to ITER, which is about half the total surface of the first wall.
Measurement of residual stresses in simplified NHF First Wall Panel prototype during post-HIP processes

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The Normal Heat Flux (NHF) First Wall (FW) panels are designed to withstand the heat flux from the plasma inside ITER. These components are made of beryllium tiles bonded to a copper alloy and 316L (N) stainless steel heat sink. A NHF FW panel consists of several fingers as elementary plasma facing units. This paper presents the experimental stress and deformations measured on a 10-fingers mock-up representative of the first stainless steel to CuCrZr joint. The scope of this work covered only this initial stage and is not representative of the final shape of the panel. Specifically, this paper reports on the monitoring of surface stresses at different steps in the manufacturing process of a simplified 10-fingers prototype of a NHF FW panel: 1) after the HIP process 2) after the solution annealing treatment and precipitation hardening of the CuCrZr layer and 3) continuous surface stress measurement during the cutting of the fingers by EDM. For this, a bespoke extensometry system has been developed. Results show that surface stress values after HIP are low, in the range of 150 MPa (Von Mises), with infrequent peaks of 300 MPa. Results also show that solution and precipitation hardening do not affect in general these values, although a slight increase in some points was observed sometimes. During the cutting operation important changes of the surface stress have been observed (average above 400 MPa), which eventually lead to visible deformations in the part. Results show that deformation is due to the release of internal stresses as new free surfaces are created. No relationship should be expected between the finger manufacturing sequence and stress. The stress release is general and affects the whole geometry of the part. Research concludes that internal stresses should be measured after the HIP process to verify the conclusions.
Analysis results of the EHF FW panel’s elements

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In the framework of PA realization, specialists from NIKIET and Efremov Institute are developing a design of First Wall (FW) Full Scale Prototype (FSP) in order to demonstrate its manufacturability and qualify critical technological processes. Design of FW FSP is developed based on the FW 14 type A. The semi-prototype has been manufactured in order to verify the FW design. Based upon the manufacturing experience, some modifications have been implemented into FSP design in order to simplify assembly and manufacturing processes. Each FW includes plasma-facing components (FW fingers), FW beam, mechanical attachment system and electrical connection system with the shield block. All the FW fingers are attached to FW beam by two poloidal straight welds. In order to provide access of welding equipment, the 60 mm slot is performed on the FW front surface. The central slot insert (CSI) is implemented into FW design to protect central slot from plasma radiation. The fixation of CSI is provided by three ring welds and the hydraulic connection is provided by two pipes. The basic design of CSI has been modified in order to simplify manufacturing and final assembly. This paper presents results of thermal and structural analysis of modified FW design in particular CSI.
P2.106

Design, analysis and manufacturing of Enhanced First Wall Panel Electrical strap

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The JSC NIKIET is responsible for the manufacture of the First Wall (FW) beam, the fingers bodies, the mechanical attachment system and electrical connection system of the FW panel to the shield block (SB) in the framework of Procurement Arrangement 1.6.1.RF.01 dated 14.02.2014. The Electrical strap (ES) is located on the FW rear surface and used for providing current through the FW to the SB and further to the Vacuum Vessel with a goal to reduce the current loops and consequently to decrease the electromagnetic loads acting on the FW during plasma disruption. The NIKIET specialists developed the new design of the electrical strap which has been implemented into the FW design. EDM cutting is used for performing the 1.2 mm thickness lamellas in the solid blank. Also a bimetallic pedestal has been added into the FW rear surface in order to provide the required electrical and thermal contact between ES and the FW. The bimetallic pedestal with 3 mm CuCrZr layer has been performed by weld deposition. This paper presents the design description of the EHF FW ES and FW bimetallic pedestal, results of thermal and structural analyses and first experience of mockup manufacturing that were performed by JSC NIKIET specialists in 2015.
P2.107

Radiative heat exchange of plasma-facing components

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The heat flux on plasma-facing components in ITER, and even more so in the projected DEMO reactor will reach values in the order of several Megawatt per square meter. Evacuating this heat in a reliable manner is key to the robustness and safety of operation of any fusion reactor. The current state-of-the-art for cooling plasma-facing components relies on cooling a high heat-resistant structure using fluids. The wall and associated cooling tubes are usually an aggregate of different materials, which by virtue of their different thermal expansion coefficients, create differential elongation stresses at elevated temperature. In addition, heat diffusion through the thickness of the materials creates a thermal barrier. These effects limit the heat flux that may be evacuated reliably. A design alternative is presented featuring a thin plasma-facing heat-resistant sheet wall which collects the heat from the plasma and rejects it to a structurally unconnected pipe forest behind the sheet wall which is at a far lower temperature. Heat exchange between the two components is through radiative heat exchange. Since the two components are structurally distinct, there is a greater freedom of choice in choosing the materials and no differential thermal stresses arise. A review of the possible designs and material choices is presented.
Thermal-hydraulic design of water cooled first wall of the fusion reactor under DEMO conditions

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The heat loads on the First Wall (FW) of the European DEMO are not yet defined, but when extrapolated from ITER, the loads can be quite high. As the DEMO will use Eurofer 97 as the structural material and Pressurized Water Reactor (PWR) conditions at the inlet, i.e. 15.5 MPa and 285 °C, the design of the heat sink gets complicated as the thermal conductivity of the heat sink material is quite low as well as the operating window of temperature (between 285 °C and 550 °C). As in ITER, there are two different kinds of heat sinks that have to be designed for the FW: the first one is normal heat flux channel and the second one is enhanced heat flux channel. For handling normal heat fluxes round tubes with counter current flow are chosen while for handling enhanced heat fluxes channel with the Hypervapotron configuration is chosen. Simulations were carried out to find out the limits from the thermal-hydraulic point of view using commercial Computational fluid dynamics code STAR-CCM+. For normal heat flux channels after optimization, the heat flux limit was found to be around 1.7 MW/m² before reaching limits from the temperature point of view (550 °C for Eurofer). And for enhanced heat flux channels the same code is used where the boiling model used was previously validated with different Hypervapotron configurations. It was found that after optimization the heat sink cannot handle heat fluxes greater than 3 MW/m² from thermal-hydraulic point of view before reaching the temperature limits of operation. Based on the results the operating limits of the different heat sinks at PWR conditions and Eurofer material as heat sink is assessed. This work was carried out within the EUROfusion Breeding Blanket Project.
Heat transfer enhancement of the DEMO first wall water cooling

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The first wall cooling of the fusion power reactor DEMO is an important part of the fusion power plant development. A cooling ability at high heat flux conditions will affect a lifetime period of the first wall modules having a direct impact on the operating costs of the fusion power plant. According to current knowledge, the water cooling provides the largest ability to remove the high heat flux using various heat transfer intensification methods. In the paper, the DEMO first wall water cooling on PWR parameters with single phase turbulators is analyzed. Effects of the intensification methods and their impact on the wall material and coolant temperatures are evaluated using thermo-hydraulic CFD calculations. The heat transfer enhancement and pressure loss of the helical fins, standard twisted tape, twisted tape with the central tube, wall detached twisted tape, standard screw tape, wall detached screw tape, dashed screw tape, solid center, and combined helical fins with the screw tape are compared. As the result, the dashed screw tape provides the best ratio of the heat transfer enhancement versus the pressure losses increase.
Design of the divertor targets shielding frame of the HEL-CZA experimental complex

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Based on the requirements of F4E, an experimental device HELCZA (High Energy Load Czech Assembly) was designed for high heat flux cyclic loading of plasma-facing components of the ITER reactor, primarily for testing of the full-size first wall modules and divertor inner vertical targets. Testing is carried out by a high power electron beam heating, and a deviation of the heat flux density at any point of the heated area must not exceed $\pm 5\%$ of the target value for the heated area. Due to Gaussian distribution of energy across the electron beam, the required high flatness requires irradiation of the area edges by the electron beam core, and the surface outside the heated area must be protected by the windows-type shielding frame absorbing approximately half the power of the beam, while the frame structure must be thin and flexible. This paper brings a design of the HELCZA shielding frame for the divertor inner vertical targets irradiation by the electron beam with a power density of 40 MW/m\textsuperscript{2}. The design includes the choice of the geometric shape of the frame, determines the frame heat load, and provides thermohydraulic calculations for the frame cooling including the determination of pressure losses and heat removal at full power failure (blackout). The design is accompanied by a basic description of the frame manufacturing process.
The paper deals with optimal electron beam heat distribution on the HELLCZa experiment calculating the flatness of the distribution of heat input and distribution of surface temperature of various samples. A computer program has been developed for balancing the heat flux in the construction materials of the sample. The first boundary condition for this calculation were primarily functions describing the process of heat input from the electron beam. The nature of this power is given power distribution beam via a Gaussian function and a screening step on the irradiated surface. The second boundary condition is the cooling of the material by the refrigerant cooling ducts. While the heat input from the electron beam is relatively flat, the distribution of subsurface cooling channels and the design of individual elements of the sample leads to significant distortions of the distribution of surface temperature of the sample material. Depending on the method of scanning and relative position of the beam to the cooling a complex problem of heat conduction emerges which must be solved numerically. The computer program solves the above problem and balances the given system of equations of heat conduction and can thus describe in detail the nature of the surface temperature of the sample at a temperature below its surface. The outputs are temperature profiles during heating, a comprehensive view of the temperature distribution in the sample and temperature profiles in selected areas below the surface of the sample. Due to good thermal conductivity of structural materials, low heat capacity and a sufficiently long time of irradiation we treated the task as stationary. As part of the solution the sample surfaces scanning by electron beam was optimized. Thus, an optimal route of the electron beam was suggested for the HELLCZa sample.
Commissioning phase of high heat flux test facility HEL-CZA

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Commissioning phase of high heat flux test facility HELCZA R. Jílek, J. Prokůpek, P. Gavila. Centrum výzkumu Řež s.r.o. (CVR), Hlavní 130, 25068 Husinec-Řež, Czech Republic, bFusion for Energy, Josep Pla, 2, Torres Diagonal Litoral B3, 08019 Barcelona, Spain *Corresponding author: e-mail: Richard.Jilek@cvrez.cz, phone: +420 601 315 137 The high heat flux test facility HELCZA is entering the commissioning phase where full parameters of the facility shall be validated. HELCZA is designed for the cyclic heat loading of plasma-facing components for ITER with heat flux densities of several MW/m² and the facility itself is capable of reaching up to GW/m² scale. The heat flux is assured by an 800 kW electron beam gun emitting electrons at 55 kV acceleration voltage. The main validated parameters are not only the maximum power of electron beam but the whole cooling system which enables to set the inlet cooling water temperature between 25 °C and 320 °C in an adjustable water pressure range up to PWR conditions, i.e. 15 MPa. The validated flow rate in the whole temperature and pressure range is adjustable up to 40 m³/h. The sample movement is assured by three types of sample holders: a) for first wall panels, b) for in-vertical target divertor and c) for flat dummy samples and small-scale samples. Besides that there is an external electromagnetic system installed enabling to change the electron beam path to ensure the required angle of incidence on the sample's surface. All the above mentioned systems have to undergo full validation during commissioning within 2016. The work leading to this publication has been funded partially by Fusion for Energy under the Contract F4E-OPE-319 and MŠMT. The opinions expressed are those of the CVR's only and do not represent Fusion for Energy's official position.
Model-based optimization of the heat flux distribution of IR-heaters for high heat flux testing

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For the testing of helium cooled plasma facing components in HELOKA-HP homogeneous surface heat flux densities of up to 500 kW/m$^2$ have to be reproduced. It has been proposed to use infrared radiation heaters which consist of several quartz glass (fused silica) tubes with tungsten filaments inside to generate the heat flux. This paper presents a numerical model of the latest type of heater which has been investigated in the SIRHEX (“Surface Infrared Radiation Heating Experiment”) facility at KIT. The model uses a transient simulation to assess the heat flux distribution on the surface of the test mock-up. It is compared with the latest results from SIRHEX which show that the newest type and set-up of heaters can produce at least 300 kW/m$^2$ for a full run of an ITER-like load (3000 cycles with 400 s plateau and 600 s dwell time) without significant signs of degradation. With the experimentally confirmed model the heater set-up will be optimized for a homogeneous distribution also for higher heat loads for future tests of plasma facing components.
Design options to avoid deep cracking of tungsten armor at 20 MW/m²

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Maintenance of structural integrity under high-heat-flux (HHF) fatigue loads is a critical concern for assuring the reliable HHF performance of a plasma-facing divertor target component. Loss of structural integrity may lead to structural as well as functional failure of the component. Currently, a full tungsten divertor was chosen by ITER Organization, and plenty of HHF qualification tests have been conducted. The tested prototypes showed that the tungsten monoblock armor often suffered from deep cracking, when the applied HHF load approached 20 MW/m². The deep cracks were initiated at the armor surface and grew toward the cooling tube in the vertical direction. The deep cracking seemed not to affect the heat removal capability of tungsten divertor, as most of the cracks were perpendicular to the loading surface. However, the inherently unstable nature of brittle cracking may likely increase the risk of structural failure. Understanding the cracking mechanism is therefore of essential importance for divertor design. In the previous work [1], a two-stage modeling approach was employed where deep cracking was thought to be a consecutive process of crack initiation and growth, which was assumed to be caused by plastic fatigue and brittle fracture, respectively. This theoretical interpretation fitted quite well with the experimental observation and revealed that the key factor for deep cracking was the plastic strain accumulation in the tungsten armor, which seemed hardly avoidable. In this contribution, designs of tungsten divertor with different dimensions and castellation based on an ITER-like divertor are proposed to avoid deep cracking at 20 MW/m². The mechanical and fracture behavior are assessed with the aid of finite element simulations. Plastic fatigue and brittle fracture failures in tungsten block are evaluated. The feasibility of manufacturing is also discussed. [1] M. Li, J.-H. You / Fusion Engineering and Design 101 (2015) 1–8
Validation of TOKES vapour shield simulations against experiments in the 2MK-200 facility

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Transient heat fluxes onto the tungsten divertor targets during disruptions in ITER may cause severe melting, leading to intolerable damage. However, for sufficiently energetic transients, tungsten vaporized from the target in the initial stage of the heat pulse will generate a protective plasma shield in front of the target, greatly reducing the incoming heat flux. This vapour shielding is a complex process, combining MHD convection and diffusion of the plasma shield with conversion of the transient heat flux into radiation. It can only really be modelled by numerical simulations, which have been performed for ITER disruptions using the TOKES fluid plasma code, demonstrating significant heat flux mitigation\cite{pestchanyi2016}. Given the potential benefits of vapour shielding with regard to the damage which may be caused by unmitigated disruptions on ITER, these TOKES simulations require experimental verification. Although plasma conditions pertinent to high energy ITER disruptions (peak heat fluxes of tens of GWm\textsuperscript{-2} on ms timescales) cannot be created easily in the laboratory, some experimental data does exist from the 2MK-200 magnetic cusp facility, where tungsten vapour shielding has been observed under relevant heat fluxes (\textasciitilde 100GW/m\textsuperscript{2}), but on much faster timescales (tens of ms)\cite{pestchanyi2016}. This device provides plasma flow over 20 ms onto a tungsten target with measured plasma density $n_e=\left(1.5-2\right)\times10^{22}$ m\textsuperscript{-3} and temperatures of $T_e=250$eV and $T_i=800$eV. This paper will describe the first ever simulation of this 2MK-200 transient vapour shield, using the TOKES code configured to match the magnetic configuration to the 2MK-200 cusp and the plasma parameters. The results are in reasonable agreement with the measured maximum density, the plasma shield width and the $T_e$ profiles. [1]S. Pestchanyi et al, accepted for publication in Fusion Engineering and Design [2]Arkhipov, et al, Plasma Phys. Rep. 20 (1994) 782
Actively cooled plasma-facing components and coolant removal system in KSTAR

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This paper deals with the first commissioning of active cooling system for plasma-facing components (PFCs) and coolant removal system. During 2015 KSTAR campaign, we have achieved a 55 sec long pulse H-mode. However, some plasma shots were terminated, not because of instabilities or limitation of heating power, but because of safety limit applied to the PFC temperature: upper boundary to lock the system is 400 °C to protect the machine, and unlocked at 200 °C. In order to overcome this limitation to achieve longer pulse, an active cooling system is installed. For commissioning, coolant was supplied into the cooling line with 14.0 l/s of mass flow using vacuum vessel cooling system. Temperature of PFCs are monitored by arrays of 200 thermocouples installed at different poloidal and toroidal locations around the torus with a time resolution of 1 sec. With the active cooling, the temperature of lower divertor is about 30 °C lower than that of upper divertor and returned quickly to initial temperature resulting in reduction of shot interval: Inertial cooling takes much longer time to unlock the safety inter-lock. Note that the capavity of current cooling system is about 24.0 l/s and not enough for entire PFCs at this stage: the coolant was supplied only at lower diveror and ploidal limiter sector except of inboard limiter, passive stabilizer, neutral beam armor. One technical issue of this active cooling of PFCs is to remove coolant from the “cooling line”. The cooling line, pipe system inside the PFCs are also used to bake the PFCs. Therefore, a coolant removal system is assembled. The system consists of adsorption dryer (external heater non purge air dryer), gas cooler, pipes and valve components. The flow capacity of adsorption dryer is 3,000Nm²/hr. The type of gas cooler is shell and tube/plate fin, capacity is 715 Mcal/hr.
P2.117

Analytical comparison of mock-ups with swirl and smooth tube for KSTAR divertor

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The tungsten (W) brazed flat type mock-up with swirl tube which consists of W, OFHC-Cu (oxygen-free high conductive copper) and CuCrZr alloy has been designed for KSTAR divertor in preparation for KSTAR upgrade. The mock-ups are tested for several thousand thermal cycles with absorbed heat flux up to 8 MW/m\textsuperscript{2} for 20 sec duration at KoHLT-EB in KAERI. In this paper, for comparison of two types of mock-ups with swirl and smooth, the hydraulic thermo-mechanical analysis is performed by using ANSYS WORKBENCH 16.0 and experimental results are compared with the Finite element analysis (FEA) prediction. FEA is performed to compare the cooling effect of the mock-ups with swirl and smooth tube and investigate the expected fatigue lifetime. To compare the cooling effect, the temperature profile of the mock-ups and the convective heat transfer coefficient of the coolant are analyzed varying the absorbed heat flux from 1 to 8 MW/m\textsuperscript{2}. To predict the fatigue lifetime of the mock-ups, the equivalent total strains calculated by the mechanical analysis are applied to the experimental fatigue curves for CuCrZr-IG. As a result, it is considered that in case of smooth tube, the local area in the water is in the subcooled boiling regime, and the cooling effect of mock-up with swirl tube is better than that with smooth tube.
The preliminary conceptual design study on the Korean fusion demonstration reactor (K-DEMO) tokamak consists of the vacuum vessel, the in-vessel components, and the superconducting magnet system, and so on [1]. The K-DEMO superconducting magnet system contains 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 magnetic fields induced by these coils enormously influence the design of divertor modules as the form of electromagnetic (EM) loads. EM loads caused by the major disruption, the vertical disruption event of plasma or the magnet fast discharge are one of the most severe external force for divertor modules as well as the thermal loads caused by high heat flux on the divertor target. The aim of this study is to estimate the EM loads on the K-DEMO divertor module by EM analyses using ANSYS-EMAG. The conceptual model of the K-DEMO divertor module including outboard and inboard targets, dome, and the cassette body with connecting supports were developed to carry out EM analyses. A water-cooled divertor concept applying tungsten monoblock type has been primarily considered [2]. Since the reduced activation ferritic martensite (RAFM) steel has been taken into account as the heat sink material for the divertor target, Maxwell force caused by the magnetization of RAFM was also estimated as well as Lorentz force induced by the magnetic field and eddy current. [1] Keeman Kim et al., “Design of K-DEMO for Near-Term Implementation”, Nuclear Fusion 55 (2015) 053027. [2] K. Im, J. S. Park, and S. Kwon, “A Preliminary Development of the K-DEMO Divertor Concept”, Submitted to IEEE Transactions on Plasma Science, 2015
A preliminary study on the rigorous 2-step (R2S) based shutdown dose rate calculations has been performed for the Korean fusion demonstration reactor (K-DEMO) in the vicinity of an equatorial port area using the coupled transport and activation calculation codes of MCNP6 and FISPACT. For the shutdown dose rate calculation, the equatorial port structures and port plug including shielding blocks were integrated into the modified equatorial port of the previously developed K-DEMO neutronic analysis model [1,2] using the CAD (Pro-Engineer™) and Monte Carlo Automatic Modeling (MCAM) programs. Then, the neutron fluxes, nuclear heating induced by neutrons and secondary gammas, and shutdown dose rate with cell-based R2S method have been calculated in all the equatorial port components. The shielding calculation by changing its thickness has also been performed to provide adequate neutron and radiation shields to protect toroidal field (TF) coils and to reduce the dose level below 100 mSv/h at 10^6 s after shutdown in the equatorial port where the personnel access for the maintenance is required. The analysis results of preliminary shutdown dose rate of all the components in the equatorial port area exceed the design target value of 100 μSv/h due to the temporally applied shielding without iterative calculations. In order to deal with this, iterative shielding calculations by changing its thickness are currently under investigation. References: [1] J. S. Park, S. Kwon, K. Im, K. Kim, T. Brown, G. Neilson, Pre-conceptual design study on K-DEMO ceramic breeder blanket, Fusion Engineering and Design 100 (2015) 159-165. [2] J. S. Park, K. Im, and S. Kwon, “Development of the Advanced Neutronic Analysis Model for the K-DEMO with MCNP Code”, SOFE-2015 (Austin, USA, May 31 - June 4, 2015) SP1-3 unpublished.
Calculations on plasma radiation heat distribution on the first walls of the K-DEMO reactor

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A pre-conceptual design study for the Korean fusion demonstration tokamak reactor (K-DEMO) has been initiated in 2012. K-DEMO is characterized by the uniqueness of high magnetic field (\(B_{T0} = 7.4\) T), major and minor radii of 6.8 m and 2.1 m, and steady-state operation. The heat load distribution by plasma radiation onto the first walls of the in-vessel components is one of the basic inputs for the various analyses including the structural and thermohydraulic analyses for further study on the K-DEMO in-vessel components. The methodology and results of calculation on the radiation heat load are presented in this paper. The KDEMO\_HEATLOAD code was developed for the calculation in a 3-D toroidal space. The plasma region is divided into 100 and 15 segments in poloidal and radial directions, respectively, to play the role of each radiation source of the core plasma. The contributions from individual radiation sources to the segmented first walls, 50 and 32 segments for the divertor and the blanket first walls, respectively, are collectively calculated considering the 3-D toroidal geometry. The K-DEMO reference power scheme with the plasma heating power of 560 MW is used for the calculation. With the radiation power ratios of \(~40\%\) and \(~90\%\) for the core and divertor plasmas, respectively, the maximum radiation heat loads on the blanket and divertor first walls were \(~0.5\) MW/m\(^2\) and \(~1.2\) MW/m\(^2\), respectively.
Detailed design and analysis of Wendelstein 7-X scraper element

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The modular stellarator Wendelstein 7-X in Greifswald (Germany) successfully started operation in 2015 with short pulse limiter plasmas. In 2017, the next operation phase (OP) OP1.2 will start once 10 uncooled test divertor units (TDU) with graphite armor will be installed. The TDUs allow for plasma pulses of 10 s with 8 MW heating. OP2, allowing for steady state operation, is planned for 2020 after the TDUs will be replaced by 10 water cooled CFC armored divertors. Due to the development of plasma currents like bootstrap currents in long pulse plasmas in OP2, the plasma could hit the edge of the divertor targets which has a reduced cooling capacity compared to the central part of the target tiles. To prevent overloading of these edges, a so-called scraper element can be positioned in front of the divertor, intersecting those strike lines that would otherwise hit the divertor edges. As a result, these edges are protected but as a drawback the pumping efficiency of neutrals is also reduced. As a test an uncooled scraper element with graphite tiles will be placed in two out of ten half modules in OP1.2. A decision to install ten water cooled scraper elements for OP2 is pending on the results of this test in OP1.2. To monitor the impact of the scraper element on the plasma, Langmuir probes are integrated in the plasma facing surface, and a neutral gas manometer measures the neutral density directly behind the plasma facing surface. Moreover, IR and VIS cameras observe the plasma facing surface and thermocouples monitor the temperatures of the graphite tiles and underlying support structure. This paper describes the integration of the scraper element and its diagnostics in Wendelstein 7-X.
P2.123

**Structural analysis of the W7-X cryopump during the superconducting coil fast discharge event**

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The cryopump will be installed for the high power and long pulse operation up to 30 minutes of Wendelstein 7-X (W7-X). The cryopump system plays a critical role for capturing ash particles from the plasma, including hydrogen, deuterium and even helium. In total there are 10 independent cryopumps, one cryopump for each of the 10 discrete divertor units. The cryopump is located along the pumping gap in the toroidal direction below the divertor target modules and close to the location where field lines hit the divertor target surface. Each pump has a design envelop of ~400mm diameter and ~4m length, and is fixed to the plasma vessel by 8 supports. Three independent cooling loops of water, LN2 and LHe, respectively, supply the cooling requirements at different conditions for its operation. During a fast discharge event of the W7-X superconducting coils, the changing magnetic field will induce eddy currents on the metallic cryopump components, especially on the low resistance copper plates and baffle fins. Mechanical forces and moments will be applied on the cryopump structure as a result of the combination of the eddy current and the background magnetic field. In order to check the impact of these mechanical loads on the structure of the cryopump, a series of electromagnetic and mechanical analyses has been performed to find out the field from the superconducting coils and plasma, the eddy current and the Lorentz force on the different cryopump components.
Local copper coating of the connectors of the divertor target elements of Wendelstein 7-X

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The 890 target elements of the high heat flux divertor of Wendelstein 7-X (W7-X) are made of a CuCrZr copper alloy heat sink armored with carbon reinforced carbon (CFC) NB31 tiles. Connectors with an internal diameter of 10 mm are electron beam welded to heat sink for the water inlet and outlet. They are produced by electron beam welding thicker tubes of CuCrZr and stainless steel with a Nickel 270 transition, which are machined at the inside and outside to reach a final thickness of 1mm. The length of the Nickel transition is 5 mm. During the incoming inspection performed at IPP, some target elements did not pass the Helium leak test in oven. In addition one element developed a Helium leak after high heat flux testing. The level of leak was always very small in the range of $10^{-4} - 10^{-6}$ Pa l/s, and always located in the welding area between the steel and Ni-Adapter transition. The detailed analysis of the manufacturing process (material charges, production batches, inspection sheets of dye penetration and x-ray) and additional examinations (non-destructive such as x-ray, metallography) did not allow understanding the reason for the initiation of these leaks. Different options were examined, and the selected solution was the electrolytic copper coating of the transition area of the connectors. The development activities of this coating are presented. The selected copper grade CW0009A is compatible with the W7-X divertor operation. The nominal thickness and the length are 0.3 and 20 mm. After bending loads up to 2000 cycles reproducing expected operation conditions of the connectors, the metallography analyses of test pieces demonstrated a good adhesion of the coating to the connector. Based on these results, it was decided to coat the connectors of the already delivered target elements.
Summary of the production of the production of the Wendelstein 7-X divertor target elements

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The actively water-cooled target elements of the high heat flux divertor of Wendelstein 7-X (W7-X) are designed to remove a stationary heat flux of 10 MW/m² on its main area and 5 MW/m² at the end adjacent to the pumping gap. A target element is made of a CuCrZr copper alloy heat sink armored with carbon reinforced carbon (CFC) NB31 tiles. The realization of the divertor requires the production of 890 target elements of five different types. In total 19.6 m² of the divertor area is shielded with 16,000 tiles. The industrial production of the target elements by the Austrian company PLANSEE SE needed 5 years. The final successful delivery of the target elements was based on an efficient quality assurance throughout the production by the manufacturer to ensure the stability of the manufacturing process and to react adequately to any quality deviations. The quality of delivered elements was finally assessed by W7-X. The most critical issue was the reliability of the bonding between CFC tiles and heat sink; each tile was inspected by pulsed thermography just after welding onto heat sink and before delivery. The recovery of CuCrZr properties was essential; heat treatment was performed after the electron beam welding of back plate to lid to produce heat sink, of the tiles and water connectors to the heat sink. The hardness and electrical conductivity of the produced elements was measured. Helium tightness of the delivered elements was systematically tested under pressure and at different temperature in a vacuum oven by W7-X. The quality of the delivered elements was confirmed by high heat flux testing based on a statistic approach. The experience gained during the monitoring of the production will be presented and discussed. The next step is 3D-machining of the individual elements to produce the 100 divertor target modules.
Progress status of the ITER Vacuum Vessel Sectors manufacturing design thermal hydraulic performance

PORTONE, Alfredo¹; CAU, Francesca¹; CAIXAS, Joan¹; PAMPIN, Raul¹; MARTINEZ, Jean Marc²; SABOURIN, Flavien²; MARTIN, Alex²; BRIANI, Pierfederico³; ALEMAN, Agustin³; FRADERA, Jorge³; COLOMER, Clara³; MARTINEZ, Emili³; ZAMORA, Imanol³; ICHARD, Mathieu³

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The ITER Vacuum Vessel (VV) is a double wall Stainless Steel structure that surrounds the plasma. It constitutes a major safety barrier for ITER, and, because of its function, is classified as Protection Important Component (PIC). Its design and construction has to follow the RCC-MR design code rules to verify the structural integrity under electromagnetic, thermal and seismic loads. Computation Fluid Dynamic (CFD) analyses of the Irregular and the Regular Sector of the ITER VV have been launched last year with the company IDOM under ITER IO and F4E contracts. The aim of these computations is to generate reliable input data (temperature field and Heat transfer coefficients) for the subsequent mechanical analyses that have to demonstrate the structural integrity of the VV and that must be submitted to the Agreed Notified Body (ANB). The Irregular Sector differs from the Regular one in the equatorial segment of the outboard (identified as PS3), in the triangular Support and penetration pipes in the outboard bottom part (PS4). Because of that, the models of the two Sectors share the same mesh in the inboard part (PS1), outboard upper part (PS2) and the common parts of PS4. Field Joints are included in the model in order to use representative boundary conditions and avoid the generation of unrealistic hot-spots at the boundaries of the sectors. The applied heat load is the neutronic heat flux calculated with MNCP code. The data for the PS3 of irregular sector have been computed by University of Wisconsin, while data for the whole regular sector are computed by F4E. The model used for the heat flux calculations of regular sector includes a detailed representation of the VV, considering heterogeneous materials; moreover, the recent modifications of the ITER blanket system are taken into account.
Electromagnetic Analysis for the In-Vessel Transfer Lines of Neutron Activation System

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In ITER the neutron activation system deploys several foil samples close to the plasma to measure the neutron fluence and the fusion power. These samples are transferred in a pneumatic way along the tubes installed on the vacuum vessel wall. Therefore, the tubes, namely transfer lines, get eddy current induced during plasma disruption, leading to Lorentz force by interacting the background magnetic field. As the transfer lines are routed along the poloidal direction of the vacuum vessel, we considered not only the poloidal field variation, but also the toroidal field variation and the halo current for calculating the electromagnetic (EM) load. The analysis results showed that the unbearable loads take place on the tube. In order to mitigate the EM loads, the transfer lines are electrically insulated from the vacuum vessel. It was shown from the EM analysis that the electrical insulation reduces the load by one order of magnitude.
P2.128

Analysis of a portable machine for post-welding operations within the vacuum vessel of ITER

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The vacuum vessel of ITER is a paradigmatic example of a gargantuan system that can only be processed in-situ and from the inside. Its assembly implies performing post welding repair operations, including machining of welding seams following the internal surface of the vacuum vessel. The requirements for the machining operations are the following: accuracy +/- 0.1 mm; dynamic machining forces 3 kN; and speed up to 1.2 m/min. In response, authors developed a new portable milling machine which was reported in SOFT2014. Now, an in-depth analysis of its capabilities is presented. More specifically, an error budget of the use of said machine is performed to estimate the accuracy of the features machined. The different error sources are clearly identified and quantified: the machine-process interaction, the inter-referencing with respect to the vacuum vessel and the geometric errors. To obtain sound estimates of the errors, both experiments and simulations have been employed. Results show that this portable machine can perform both mid-duty milling and drilling operations in a five axis configuration. Therefore, it is not only suited for welding seam recovery, but also drilling and milling of biscuits and edge preparation. The expected results of using the machines have shown uncertainties of 0.06 mm, heavily determined by the machine-work piece inter-referencing method, which becomes the major improvement vector for further research around these in-situ machines. In terms of the process, tool deflection is the main error component and should be studied in detail. Also, thermal issues and deformations should be taken into account if the use of portable machines refers to continuous or intensive machining operations.
Progress on the Design Development and Prototype Manufacturing of the ITER In-Vessel Coils

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ITER is incorporating two types of In-Vessel Coils (IVCs): ELM Coils to mitigate Edge Localized Modes and VS Coils to provide Vertical Stabilization of the plasma. Strong coupling with the plasma is required in order that the ELM and VS Coils can meet their performance requirements. Accordingly, the IVCs are mounted on the Vacuum Vessel (VV) inner wall, in close proximity to the plasma, just behind the Blanket Shield Modules (BSM). Due to high radiation environment, mineral insulated copper conductors enclosed in a steel jacket have been selected. The reference design and prototype work provided a good basis for the development of radiation resistant conductor capable of operating within the harsh conditions in ITER vacuum chamber. However, this effort identified shortcomings in achieving satisfactory manufacturing solution, and most significantly, difficulties in brazing the brackets onto the ELM coil conductor. Since this process has not proven successful, alternative designs are under development and prototyping. Prototype manufacturing on the alternative designs has been completed at ICAS, Italy and ASIPP, China. The aim was to eliminate the need for internal coil joints, to prove the principle of longer conductor length manufacturing, and to perform bending and welding trials on two different conductor cross-sections: circular and square. The procurement of the IVCs and their conductors will be done via direct call-for-tenders from the ITER Organization and preparation has already started. This paper will give an overview of the alternative design and prototype manufacturing of the ITER In-Vessel coils.
Analysis and experimental justification of electrical connector for ITER blanket module

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In ITER blanket system, electrical connectors (“E–straps”, ES) are used to form a low impedance electrical path from shield blocks (SB) to the vacuum vessel (VV). Main functions of ES is providing current from SB to VV. ES shall withstand electromagnetic (EM) loads and Joule heating resulted from electrical current with magnitude up to 137 kA during 300 ms, accommodate cyclic relative displacements SB-to-VV and operate in harsh environment: neutron radiation, thermal cycling and high vacuum. In line with Technical Specification to Procurement Agreement (PA) 1.6.P3.RF.01.0 on Blanket Module Connections (BMC) the ES is manufactured by assembling of identical Z-shaped plates (lamellas) with spacers between them and thick plates at each extremity lamellas joined together by brazing or mechanically by studs ensuring good electrical contact. Manufacturing of the ES from one solid blank by electrical discharge machining (EDM) has been developed and proposed by NIKIET instead of the assembling from separate parts. EDM method enables to avoid the shortcomings inherent to the ES reference design made from formed plates. This paper presents design and analyses efforts applied on development of ES made of solid blank. Patterns of temperature, EM loads and stress-strain state, accumulated fatigue damages have been found with numerical simulations for all specified loading conditions. Strength assessment shows that current design of ES meets the applicable design code: structural design criteria for in-vessel components (SDC-IC). Results of electrical and cyclic mechanical tests at temperature are presented in this paper.
P2.131

Integration and mitigation aspects of the updated ITER ICH antenna shutdown dose rate analysis

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The Ion Cyclotron Heating and Current Drive system (ICH) is designed to launch RF power into the ITER plasma, and will reside in equatorial ports (EP) 13 and 15. Shutdown dose rates (SDDR) within the ICH port interspace are required to be ALARA and less than 100 μSv/h at 10⁶ seconds cooling, in locations where hands-on maintenance is required. The shielding performance of in-vessel, vessel and port systems are severely jeopardized by streaming paths such as the gaps between port frame and the ICH antenna. Recent design integration and assembly needs requested larger gaps to be compensated by streaming labyrinths. The impact on the SDDR levels due to these changes and further mitigation strategies have been studied in this work. An accurate description of in-vessel systems, consisting of blanket modules, manifolds and coils around the ICH port plug is needed and has been added to the previous MCNP model of the ICH antenna in the ITER torus sector reference model (C-lite). Several modified front shim designs of the gap dogleg labyrinths were trialed containing boron carbide to act as a neutron absorber. Initially average dose rates at 10⁶ seconds cooling were found to be on the order of 500 μSv/h, approximately doubled from previous analysis. However, this was primarily due to the effect of the fully open lower cryopump port, which in the previous model had been represented as a well-shielded diagnostic lower port. Replicating the shielded lower port environment in updated calculations resulted in dose rates of 235 μSv/h, a marginal reduction from the previous design. The attractive neutron attenuation capabilities by boron carbide shield elements in several locations of the ITER tokamak have been demonstrated. However, open lower ports potentially lead to significant radiation ‘cross-talk’ to the equatorial port interspace regions jeopardizing respective shielding design efforts.
Engineering analysis of the DSM and PCSS@ISS of ITER upper ports #2 and #8

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In this paper the stress-strain state of the diagnostic shield modules (DSM) and the supporting frames (ISS, PCSS), located in the upper ports #2 and #8 of the tokamak ITER is investigated. DSM is the upper port components and has two main functions: neutron radiation protection and maintenance of rigid fixation diagnostics placed in the port. DSM is operated at high temperatures, significant electromagnetic loads and intensive flux of high-energy neutrons. During the structural analysis it is required to take into account all these loading factors. DSM consists of four massive steel parts joined by welding. Cooling channels are located within these parts. For the efficiency analysis of the cooling system the thermo-hydraulic calculation using Ansys CFX system is made. The initial data for determining the volume heat generation was the result of neutron calculation. The thermal state of the DSM in normal operation is calculated and the characteristics of the coolant flow in the cooling channels are obtained. The resulting temperature field is used to determine the thermal stress state. Also in the work the calculation of dynamic electromagnetic loads in the individual parts of the structure is carried out. The presented calculations are executed by numerical simulation using ANSYS Maxwell, Mechanical. The solution of the electromagnetic dynamic problem with the plasma failure scenario VDE"Up" is gathered. The input data for this task was the result from simulation of the code DINA. We investigate the DSM and the supporting frames structural integrity under the action of seismic loads by the linear-spectral method. Based on the equivalent stress values obtained as a result of the seismic analysis, conclusions about the possibility of operating under the influence of the seismic loads are made. Based on the results of the seismic analysis the interface loadings (response spectrums) at the attachment points are calculated.
Engineering analyses of the upper vertical neutron camera of ITER

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The primary systems of future international thermonuclear experimental reactor (ITER) have to withstand major thermal, nuclear, electromagnetic and seismic loads. Therefore, engineering analysis of elements of construction plays crucial role in realizing of the project as a whole. The paper describes calculations of spatial stress-strain state from major loads arising during operation upper vertical neutron camera (VNC) – subsystem of neutron diagnostics ITER, designed for measurement of plasma neutron source with spatial and transient resolution. It is in port-plug of upper port ITER #18. Detectors of VNC located inside box type body. To provide neutron shield, original version of installation assumed to use composite structure containing tungsten plates, steel plates and granular boron carbide. According to results of thermal analysis, there was found that in this case arises the great overheating of installations body. That requires creating the complex cooling system. In this regard, there was develop new version of design, which has significant difference. The major difference is offset in depth to port from plasma, which led to decrease the neutron flux. New design consists of box type body, detector modules and steel matrix, which filling the inward space. It was calculated transient thermal analysis for determination temperature distribution of VNC during normal operation. Based on the results from thermal analyses, it was calculated thermal-structural analysis. It was calculated transient electromagnetic analysis using Ansys Maxwell to determinate dynamic loads, which arose during plasma disruptions. Obtained ponderomotive forces transferred to structural analysis for determination of stress-strain state during the most dangerous disruption events. Stress-strain state from seismic load determined using static and linear spectrum method. At finale stage, it was calculated structural analysis using applicable load combinations to construction.
P2.134

Application of fracture mechanical assessment to in-wall shield ribs in ITER vacuum vessel

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Korea has been manufacturing two vacuum vessels of ITER and main jointing method to in-wall shield assemblies is welding. Though in-wall shield ribs holding neutron shielding blocks should sustain various design loads such as electro-magnetic forces, earthquake and their own weights, as a part of the assembly, in-service inspections are hardly possible because they are installed between double-wall of the vacuum vessel. In this study, a series of linear elastic fracture mechanics assessment was carried out to examine structural integrity of the in-wall shield ribs for supplement of current design approach. Two kinds of idealized cracks were assumed and conservative assessment conditions were selected from relevant design documents. Particularly, fatigue crack growth evaluation was performed according to well-known fitness-for-service codes and their validity was checked via finite element analyses. As a result, it was proven that the in-wall shield ribs had sufficient structural margins even if fracture mechanics was applied to the plausible crack under typical loading conditions. Besides, effects of applicable codes as well as artificial crack shapes, sizes and orientations were evaluated quantitatively. The detailed assessment method could be used for subsequent structural integrity assessment of other ITER component manufactured by the welding.
Components Design and Performance Analysis for the Preliminary Design of HCCR TBM-set

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After the Conceptual Design Review (CDR), Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM) design is being updated for the preparation of the preliminary design phase. The manufacturability is considered based on the TBM-set model of CD phase. The overall geometry of the first wall, side wall and the breeding zone was changed slightly. The thermal-hydraulic and mechanical analysis are conducted with ANSYS and ANSYS CFX v14.5 on the modified model. Additionally, the detailed design work are performed on the back manifold (BM), TBM-shield and the connecting supports which are the components of the TBM-set. The internal channels for the He coolant and purge gas are designed in BM itself. The characteristics of the He channel is analyzed to meet the requirements of the temperature and the pressure drop. The design of the TBM-shield is changed in order to simplify production and assembly. The geometric design of the connecting supports are referred from the connection design of the blanket first wall. The other types of design are H-beam and a flanged connection which are typically used in the industry. The stress distribution and strain are investigated according to the loads condition.
P2.136

Multiphysics engineering analysis for high field side reflectometry

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The High Field Side Reflectometry is diagnostic equipment subjected to the conditions that are severe even for ITER: magnetic field over 9T, temperatures up to 700 °C, strongly non-uniform temperature field, specific shape of the equipment with length of in-vessel waveguides about 10m and location of waveguides close to the blanket connectors where large halo currents are expected during disruptions make the development of reliable design challenging and make the role of multiphysics simulations extremely important. This study deals with eddy current, halo current, thermal and structural analyses of HFS Reflectometry. Specifics of the equipment leads to the necessity of performing electromagnetic analysis on the basis of global FE model of ITER sector including all in-vessel HFS Reflectometry components, which was developed and benchmarked against available at ITER database results. Modified approach to halo current analysis of HFS Reflectometry is proposed and discussed. Rational models for thermal and structural analyses as well as corresponding computational procedures are proposed and principal results of multiphysics simulations are presented and analyzed.
Divertor Thomson scattering structure integrity report

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The presentation is focused on the simulation results and approaches used for loading analyses made for DTS in-vessel equipment, including spatial stress strain state, seismic analysis, electromagnetic analysis as well as the most important load combinations. Finite element model of the construction was updated according with updated DTS components design and separated on the following construction parts: outer frames of the front and back diagnostic racks, neutron shield attached to the front rack outer frame as well as front and back inner frames with optical components. The nuclear heating released in the front rack constructions was used as an input data for the transient thermal analysis, resulting temperature map behavior during normal operation mode with 500 MW of fusion power. The thermal analysis with and without taking into account black body radiation of the front rack construction outer surfaces were made to obtain upper and lower heating estimates without accounting reabsorption of the radiation. Basing on the thermal analysis results, the DTS in-vessel elements stress strain state was analyzed and the most stressed points were listed. Both linear spectrum and equivalent static methods were used for seismic analysis to obtain stress and displacement maps in construction due to seismic loads. Electromagnetic analysis was done to determine applied to the constructions forces and moments for the most severe plasma disruption events. Stress strain state and temperature maps due to plasma disruption events were obtained also. Some incident and accident events were studied being enveloped into a single event called “Accident”. This event takes into account the worst conditions of significant incident and accident events. Finally, the load combinations of event categories I-IV were studied and the obtained results were analyzed.
P2.138

Electromagnetic Analysis for HL-2M RMP Coils In Vessel

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HL-2M RMP (Resonance Magnetic Perturbation) Coils is designed to provide a resonant perturbation magnetic field for high beta plasma operation scenarios stability control, such as Edge Localized Modes (ELMs) suppression control, Resistance Wall Model (RWM) fast control and Error magnetic field correction control, etc. Especially, ELMs result in impulsive burst of energy deposition on to the “Plasma Facing Components” (PFCs), causing a reduction in their lifetime. Various experiments have shown the application of “Resonant Magnetic Perturbation” (RMPs) produced by in-vessel non-axisymmetric coils can be used to suppress ELMs. HL-2M RMP coils consist of eight toroidal sectors of two(Upper and Lower) 4-turn round coils, total of 16 coils are supported above inner wall of vacuum vessel and positioned behind the First wall, without any welding contacts between them. its location has at least 70mm distance to the plasma edge. 25mm outer diameter of mineral insulated cable has been formed as coil conductor and each conductor is about 16m length. the RMP coils are rated 190V and 2.75kA per turn, DC to 100Hz. The electromagnetic analysis results indicate RMP coil could resist plasma disruption on plasma edge, via an its invisible magnetic field; the maximum poloidal magnetic indensity is about 0.019T on plasma edge, the different RMP coil current produce the different poloidal magnetic indesy on plasma edge. The different electrical current or phase on difference sector RMP coils could produce toroidal magnetic vector, it perhaps could contribute to the plasma rotation scenarios. Plasma initial current 3MA decline to 0 by lineal analysis for disruption Senario,while plasma moving from its initial position (R=1.78m) to the edge(R=2.4m), maximum eddy current on RMP coils is estimated and electromagnetic force is about 15kN/m, the maximum stress on RMP coils is about 35MPa, smaller than allowable value. KEY WORDS:HL-2M RMP Coils, Electromagnetic Analysis, Disruption Senario
P2.140

Thermal radiation analysis of DEMO tokamak

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Thermal radiation analysis of the DEMO tokamak based on the updated CAD design of in-vessel components and magnet system has been carried out. For the purpose of the analysis, Vacuum Vessel Thermal Shield (VVTS), Cryostat Thermal Shield (CTS) and some support structures have been created additionally (on a conceptual level) to complement the overall DEMO CAD design model. The Finite Element (FE) code ABAQUS was used to perform numerical analyses. Thermal radiation simulations in complex geometries can be largely affected by numerical errors due to calculation of geometrical view factors. Hence a special care was taken to adapt the geometry and mesh of the FE model to reduce the energy flux error to an acceptable level. The main simulation results provide thermal loading on different DEMO systems and components that enable also the calculation of refrigeration power required to cool the magnet systems and thermal shields. In addition, different thermal shielding configurations and scenarios were considered. Besides the base case with actively cooled thermal shields, also the configurations with passive Multi Layer Insulation (MLI) were analysed. The study also demonstrates the importance of 3D simulations for optimisation of thermal shielding in order to reduce the local heat load peaks.
P2.141

Structural analysis of demo divertor cassette body and design study based on rcc-mrx

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This study is a part of the structural activity being conducted in the framework of the structural design of a DEMO Divertor. The thermal and structural analysis has already been started since a year and the first results has been partly published in a previous paper. The Cassette Body is being analyzed considering the most critical types of loads (e.g. coolant pressure, volumetric neutron heating and EM loads) according to their latest estimates. This work is based on the design-by-analysis approach adopted in the conceptual design study of the DEMO Divertor. The divertor design has been assessed in terms of a number of variables e.g. loads, key geometric dimensions, positions of the Cassette attachments to the vacuum vessel, or positions of load application, in order to enhance the knowledge about the structural behavior of the Divertor Cassette. In addition to the existing 3D solid element model, also a shell element model has been developed: so that an extensive parametric analysis can be easily done for a comparative estimation. The structural assessment was done according to the Design and Construction Rules for Mechanical Components of Nuclear Installation (RCC-MRx).
Propagating decay gamma source uncertainties to dose rate in SDR calculations with R2SUNED

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Shutdown dose rate (SDR) analysis plays a key role in the design of fusion facilities like ITER and DEMO. One of the most used methodologies to carry out SDR calculations is the rigorous-two-step (R2S) method based on the coupling of transport and activation calculations. Currently, one of the most relevant lacks of this method is the possibility to propagate the effect of the uncertainties accumulated along the calculation process to the error in the final dose evaluation. This error propagation represents an active and relevant frontier in neutronics research. In SDR calculations using computing tools based on Monte Carlo (MC) transport simulations, the most important source of uncertainty is the statistical error. In SDR calculation statistical uncertainty arises from neutron and decay transport simulations. Thus, in R2S methodology, the neutron flux uncertainties must be propagated to the decay gamma source, and then the error on decay gamma source has to be propagated consistently with statistical uncertainties due to the gamma transport process to the final dose value. In this work, the scheme implemented in the R2SUNED tool to propagate uncertainties from the decay gamma source to final dose is described and applied. This method uses the knowledge of the contribution of each phase space element of the decay gamma source (i.e., the contribution of each energy group of each emitting volume) to the final dose to propagate the decay source error to the dose error value. This scheme has been applied to the ITER benchmark exercise in order to check its applicability to problems where the decay source has a large spatial extension (requiring a large number of mesh elements). In this exercise the decay gamma source were obtained using a direct-one-step method which can provide both decay source intensities and their relative errors due to the neutron flux statistical uncertainty.
Investigation and Optimization of Molybdenum Disulfide Parameters for Low Friction Coating

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Molybdenum disulfide (MoS\(_2\)) coating was deposited by magnetron sputtering onto the target material. The coatings of deposited MoS\(_2\) can be used in high vacuum and aerospace environments for lubrications purposes, which ultra-low friction is desirable. For these reason, the sputtered MoS\(_2\) coating method is primarily considered for ITER components and their mechanical assemblies. A common deposition technique widely used is sputtering, the sputtering process is ideally suited to coat precision mechanical components with exquisitely thin, uniform thickness. It also permits friction coefficient and durability of the MoS\(_2\) coating. However, its application of sputtering to surfaces by the technique of depositions is a complex process involving many variables. Usually, since very thin MoS\(_2\) coating are used for frictional control, it is important to understand the relationship between the sputtering conditions, their friction and wear behavior. Therefore, it seems necessary to investigate the conditions and effects of MoS\(_2\) parameter to achieve the best coating having the desirable qualities. The objective of this study is to describe the understanding of coating process and obtained the optimal MoS\(_2\) parameters using the experiments design which is central composite design. To achieve the best surface quality, the task will be optimized by sputter-depositing MoS\(_2\) disc under various conditions of Argon gas pressure, initial surface finish and sputtering power. Moreover, the evaluating the measurement of friction coefficient and coating thickness were performed using a pin-on-disc apparatus.
Application of the VTTJ cold junction technique to fusion reactor relevant geometry and materials

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A new technique, called Vacuum Tight Threaded Junction (VTTJ), has been developed and patented by Consorzio RFX, permitting to obtain low-cost and reliable non welded junctions, able to maintain vacuum tightness also in aggressive environments. The technique can be applied also if the materials to be joint are not weldable and for heterogeneous junctions (for example, between steel and copper) and has been tested up to 500 bar internal pressure and up to 200°C. The main advantages with respect to existing technologies are an easy construction, a low cost, a precise positioning of the junction and a high repeatability of the process. Due to these advantages, the new technique has been adopted for several components of the SPIDER experiment and has been also recently accepted by the ITER vacuum group for the usage in the MITICA experiment, the full prototype of the ITER Neutral Beam Injectors. Recently, the VTTJ technique has been tested with geometry and materials compatible with the divertor and other components of future fusion reactors. Namely, a set of junction samples have been manufactured, joining CuCrZr to 316L stainless steel and using tube-to-tube geometry. Three different geometries of the steel part have been adopted in order to test possible alternatives that could be advantageous in terms of corrosion compatibility, especially in view of DEMO. The samples have been tested according to the ITER criteria for the qualification of the heterogeneous junctions of the ITER divertor. The main results of the test campaign are described in the paper.
On the thermo-mechanical behaviour of DEMO water-cooled lithium lead equatorial outboard blanket module

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Within the framework of EUROfusion R&D activities an intense research campaign has been carried out at the University of Palermo, in close cooperation with ENEA Brasimone, in order to investigate the thermo-mechanical performances of the DEMO Water-Cooled Lithium Lead breeding blanket (WCLL). In particular, attention has been paid to the most recent geometric configuration of the DEMO WCLL outboard equatorial module, as designed by WCLL project team during 2015, endowed with an attachment system based on the use of radial pins, purposely outlined to connect the module back-plate to its back-supporting structure, that have been properly considered to simulate more realistically the module thermo-mechanical behaviour. The research campaign has been mainly focused on the investigation of the module thermo-mechanical performances under the Normal Operation (Level A) and Over Pressurization (Level D) steady state loading scenarios envisaged for the DEMO WCLL breeding blanket. A theoretical-numerical approach, based on the Finite Element Method (FEM), has been followed and the qualified ABAQUS v. 6.14 commercial FEM code has been adopted. Thermo-mechanical results obtained have been assessed in order to verify their compliance with the design criteria foreseen for the structural material. To this purpose, a stress linearization procedure has been performed along the most critical paths located within the module structure, in order to check the fulfillment of both Level A and Level D rules prescribed by the RCC-MRx structural safety code. Results obtained are herewith presented and critically discussed, highlighting the open issues and suggesting the pertinent modifications to DEMO WCLL outboard equatorial module design aimed to obtain the complete fulfilment of the prescribed safety criteria.
Dynamic thermal-hydraulic modeling of the EU DEMO breeding blanket cooling loops. Part II: WCLL

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The development of a system-level thermal-hydraulic model of the whole EU DEMO tokamak has been launched by the EUROfusion Project Management Unit. In order to follow the progress in the design of the tokamak components, the model should be developed in an object-oriented fashion, to ensure a high modularity. Within this framework, the first block of the model is under development at Politecnico di Torino, to simulate the breeding blanket cooling system. The breeding blanket block of a large thermal-hydraulic model, other than being a fundamental link between the in-vessel heat source and the ex-vessel power conversion systems, can provide valuable elements to compare different coolant options, as well as different cooling scheme structures currently investigated inside the Breeding Blanket Work Package. Moreover, the model will provide suggestions to increase the heat removal efficiency of the cooling system. While the module related to the Helium-Cooled Pebble Bed (HCPB) blanket concept has already been developed and presented, this paper focuses on the model for the Water-Cooled Lithium Lead (WCLL) blanket concept, which is currently under design at ENEA Brasimone. The lumped-parameter transient model of the WCLL concept, developed using the Modelica language, is presented, and the results under steady-state loads are reported. Then, the thermal-hydraulic performances of the designs of the WCLL and HCPB as implemented into the blanket thermal module are compared for the same operating condition, in terms of the total pumping power required to keep the structures at a safe temperature.
The Breeding Blanket is a key component in a fusion power plant in charge of ensuring tritium breeding, neutron shielding and energy extraction. Water-Cooled Lithium-Lead Breeding Blanket (WCLL) is considered a candidate option in view of the risk mitigation strategy for the realization of DEMO. Indeed, this design might benefit of efficient cooling performances of water as coolant, as well as of a power conversion system, based on conventional and reliable balance of plant. ENEA CR Brasimone has developed during 2015 a design of the equatorial outboard module based on horizontal (i.e. radial-toroidal) water cooling tubes in the Breeding Zone (BZ), and on PbLi flowing in radial-poloidal direction. Therefore, besides the caps zone, the module is composed by 14 segments having the same geometry. A CFD model by ANSYS CFX-15 is developed to investigate the thermal-hydraulic efficiency; to evaluate the temperature distribution in the structures and the thermal field and flow path in the breeding zone. The mesh models the breeding zone, the water cooling system in the breeding zone and in the first wall and the metallic structures. Periodicity conditions are imposed at the upper and lower horizontal stiffening plates. Sensitivity analyses are performed to optimize the layout of the design, thus enhancing temperature distribution in the module. Results show a margin of 30°C from the maximum allowed temperature of the EUROFER. The maximum thermal load tolerable by the first wall is also evaluated.
Thermal-Hydraulics CFD analysis of WCLL BB manifold

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Within the framework of EUROfusion Power Plant Physics & Technology Work Programme, the Water Cooled Lithium Lead (WCLL) is one of the four breeding blanket (BB) concepts considered as possible candidate for the realization of DEMO fusion power plant. ENEA CR Brasimone has developed during 2015 a new design of the outboard module based on horizontal (i.e radial-toroidal) water cooling tubes in the Breeding Zone (BZ), and on PbLi flowing in radial-poloidal direction. Radial-toroidal and radial-poloidal stiffening plates define the PbLi flow pattern; a gap between the Back Plate (BP) and the BZ constitutes the PbLi inlet manifold. Orifices in the radial-poloidal stiffening plates and in the plate facing the BZ ensures the PbLi distribution in the BZ. A conceptual design of the new WCLL BB has been analyzed to investigate the behaviour of PbLi in the manifold region and to optimize the mass flow rate distribution in the BZ of the module. Different geometries have been analyzed, considering orifices of different dimensions. The optimal solution is a compromise between the need to have slow PbLi velocity, to limit Eurofer corrosion, and to preserve the structural capability of the stiffening plates to withstand the overpressure conditions. The results of the analytical solution are analyzed by a detailed three-dimensional CFD analysis using ANSYS CFX-15 code.
CFD simulation of the magnetohydrodynamic flow inside the WCLL breeding blanket module

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The interaction between the molten metal and the plasma-containing magnetic field in the breeding blanket of a Tokamak fusion reactor causes the onset of a magnetohydrodynamic (MHD) flow. In order to properly design the blanket, it is important to quantify how and how much the flow features are modified compared with an ordinary hydrodynamic flow. This paper aims to characterize the evolution of the fluid inside one of the proposed concepts for DEMO, the Water-Cooled Lithium Lead (WCLL), focusing on the central cell of the equatorial outboard module. The study was carried over with the CFD code ANSYS CFX-15. A preliminary validation was required in order to gauge the capability of the electromagnetic model employed by the code to deal with MHD problems. The buoyant and pressure-driven fully developed laminar flows in a square duct were selected as benchmarks and the numerical results were then compared with theoretical solutions. An excellent agreement was found for all the test cases investigated. The channel analysis was realized on a simplified version of the latest available design geometry, developed by ENEA CR Brasimone and its associated partners in the framework of the EUROFusion Power Plant Physics & Technology Work Programme. The simulation highlighted various interesting features, including high velocity jets close to the baffle plate and the onset of an anti-symmetrical electric potential distribution. The electromagnetic pressure drops in the channel were also estimated and found consistent with previous results obtained for similar configurations. Follow-up activities will include the validation of the code for more complex scenarios and the extension of the analysis to the whole of the LiPb circuit to support the WCLL development.
MHD flow and heat transfer in model geometries for WCLL blankets

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A number of liquid metal blanket designs for applications in nuclear fusion reactors is currently under development. In the water cooled lead lithium (WCLL) blanket Eurofer97 is used as structural material and liquid PbLi as breeder, neutron multiplier, and as heat transfer medium. The released heat is removed by water at a pressure of 155 bar (pressurized water reactor conditions, 285°C - 325°C) that flows through cooling pipes immersed in the liquid-metal pool. The temperature at the Eurofer/PbLi interface has to be kept in the range between 300°C and 550°C. In order to withstand disruption-induced forces and water pressure in case of accidental conditions, the breeder zone is stiffened by internal plates that form rectangular ducts in which the liquid metal flows. Numerical simulations have been performed to predict liquid metal flow distribution in model geometries for WCLL blankets under the influence of intense external magnetic fields. The flow may be driven by an applied pressure gradient and/or by buoyancy due to the presence of volumetric heat sources in the fluid and heat transfer at the pipes. Magnetic fields of different strengths are applied perpendicular to the main flow direction and buoyant and mixed convection are studied for various thermal conditions. In the present design of WCLL blankets with cooling pipes in vertical channels, internal layers develop along magnetic field lines tangentially to the tubes. Across these layers significant velocity gradients occur.
P2.152

Structural analysis of the back supporting structure of the DEMO WCLL outboard blanket

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Within the framework of EUROfusion R&D activities an intense research campaign has been carried out at the University of Palermo, in close cooperation with ENEA Brasimone, in order to investigate the thermo-mechanical performances of the Back-Supporting Structure (BSS) outboard segment of the DEMO Water-Cooled Lithium Lead breeding blanket (WCLL). In particular, the configuration of the BSS outboard segment, purposely set-up by the WCLL project team during 2015 according to the blanket “multi-module system” concept, has been taken into account in order to study its steady state thermo-mechanical behaviour, paying attention to the simulation of both modules-BSS and BSS-vacuum vessel attachment strategies and concepts. The research campaign has been mainly intended to investigate the thermo-mechanical performances of the BSS outboard segment when subjected to both the thermo-mechanical and electro-magnetic loads it is foreseen to undergo under particularly critical steady state loading conditions. A theoretical-numerical approach, based on the Finite Element Method (FEM), has been followed and the qualified ABAQUS v. 6.14 commercial FEM code has been adopted. Thermo-mechanical results have been assessed in order to verify their compliance with the design criteria foreseen for the structural material. To this purpose, a stress linearization procedure has been performed along the most critical paths located within the highly stressed BSS regions, in order to check the fulfilment of the rules prescribed by the RCC-MRx safety code. Results obtained are herewith presented and critically discussed, highlighting the open issues and suggesting the pertinent modifications to the BSS geometric design aimed to obtain the complete fulfilment of the prescribed safety criteria.
Consistent post-test analyses of LIFUS5 experiment

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The interaction between lithium-lead and water is a major concern of Water Coolant Lithium Lead (WCLL) breeding blanket design concept, therefore deterministic safety analysis of the in-box LOCA postulated accident is of primary importance. In this framework, a past experimental campaign was carried out in LIFUS5 to investigate the evolution and the consequences of the interaction. Then, these data were used for evaluating the predictive capabilities of codes. No code was found able to simulate these experiments, which consider both the thermodynamic liquid metal-water interaction and the exothermic chemical reaction, including the evaluation of the hydrogen production, without engineering assumptions. This would imply that only conservative bounding analyses would be possible for the in-box LOCA, resulting in unrealistic predictions of the accident scenarios and in an excess of conservativeness in the system design. In view of these considerations, a modified version of SIMMER-III code (Ver.3F Mod0.0) has been set up and is under validation at UNIPI/ENEA. The paper presents the post-test analyses by SIMMER-III Ver.3F Mod0.0 of LIFUS5 Test #3 experiment carried out in the framework of the EUROfusion Project. A series of sensitivity analyses are performed to overcome uncertainties in the test data and experiment execution, and to investigate the capability of the code in predicting both thermodynamic and chemical phenomena and processes occurring during PbLi/water interaction. Pressure trends, temperature evolutions and injected mass flow rate are compared and reported. Results show agreement between numerical results and experimental data in the long term. Besides, the predicted pressure trend bounds the experimental trends (i.e. conservative results), differences are observed in the first second of the transient due challenges in mastering the complex multi-field multi-fluid phenomena occurring during the injection. Limits of simulation due to imperfect knowledge of thermo-physical proprieties of the chemical reaction products are also pointed out.
Conceptual design of the water cooled ceramic breeder blanket for CFETR

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China Fusion Engineering Test Reactor (CFETR) is an ITER-like superconducting tokomak reactor. Its major radius is 5.7m, minor radius is 1.6m and elongation ratio is 1.8. It is possible to upgrade to R~6 m, a~2 m. CFETR mission and objectives are to bridge gaps between ITER and DEMO, and to realize fusion energy application in China. CFETR has two phases. Phase I is to demonstrate full cycle of fusion energy with Pf = 200MW, and to demonstrate T self-sustained with TBR ≥ 1.0 and long pulse or steady-state operation with duty cycle time ≥ 0.3 ~ 0.5. Phase II is to perform validation of DEMO technology, including advanced blanket with TBR ≥ 1.1 and high electricity gain. As a risk mitigation strategy, three blanket concepts (i.e. the He-cooled ceramic blanket, the He-cooled LiPb blanket, and the Water-cooled ceramic blanket) are under development and evaluation in parallel for CFETR from ITER TBM program. Two options of the Water-cooled ceramic blanket for CFETR are being designed in Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP). One employs PWR condition. Another is focused on a water cooled breeder blanket with superheated steam as advanced option. It is expected to enhance tritium breeding in the viewpoint of neutronics, on the other hand, the superheated steam at higher temperature can improve thermal conversion efficiency. Li2TiO3 pebbles and Be12Ti pebbles are chosen as tritium breeder and neutron multiplier respectively for two options. In this contribution, designs and performance analyses of the water cooled breeder blanket for CFETR phase I and Phase II are reported under PWR and superheated steam cooling conditions, respectively. Main issues are proposed and discussed.
Experimental study on the subcooled boiling in square channel

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Square channel is widely used in the conceptual design of water cooled blanket of fusion reactor for cooling and providing appropriate inner temperature field for tritium breeding. Thermal hydraulic design of blanket directly determines the heat transfer efficiency and safety characteristics of fusion reactor. Therefore, thermal-hydraulic characteristics of square channel should be investigated. The appearance of subcooled boiling greatly affects flow instability of coolant channels in blanket and safety characteristics of fusion reactor. Onset of nucleate boiling (ONB) is the cut-off point of single-phase flow and two-phase flow, and the flow and heat transfer characteristics will change a lot after ONB. For 8×8 mm² square channel in blanket, the experimental study of subcooled boiling has been conducted. Experimental results showed that onset of nucleate boiling was strongly affected by system pressure and heat flux, while the effect of inlet supercooling temperature of the experimental channel was not very significant. Comparing the experimental results with Bergles-Rohsenow correlation and other correlations, it can be found that the correlations which had been developed by other researchers could not represent subcooled boiling characteristics in square channel. Therefore, a new empirical experimental correlation was developed on the foundation of experimental data. The new correlation can predict the relationship between the wall superheat and heat flux at ONB with the maximum relative error of 35%.
Thermal-hydraulic responses of WCCB to the transient operation for CFETR

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The Water Cooled Ceramic Breeder blanket (WCCB) is being researched for Chinese Fusion Engineering Test Reactor (CFETR). From the security point of view, the thermal-hydraulic analysis is very essential because the blanket should remove the high heat flux radiated from the plasma and the volumetric heat generated by neutron wall loading. For the normal state of plasma burning, the jumped peak heat flux during the transient operation can occur. Besides, for the plasma transient disruption state, including the major disruption (MD), vertical displacement events (VDEs) and edge localized modes (ELMs), the unpredictable huge heat flux can be released. Both transient states will cause serious damage on the plasma facing components, especially the first wall. The temperature of tungsten and RAFM steel will probably exceed the allowable upper limits. In this paper, based on the detailed whole computational model of WCCB, the responses of three dimensional (3D) thermal-hydraulic to the two kinds of transient operation are analyzed by using Computational Fluid Dynamics (CFD) method. The fluctuation variation of temperature with time under different transient operations is obtained, contributing to evaluate the damage degree of materials. Furthermore, the comparison of temperature contour of effected zones along the radial direction under different duration time of peak heat flux has been made.

Keywords: WCCB, transient operation, thermal-hydraulic
Activation analysis of CFETR WCCB blanket

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The water-cooled ceramic breeder (WCCB) blanket is one of the candidates of Chinese fusion engineering test reactor (CFETR). WCCB blanket will produce radioactive waste during its operation and decommissioning processes. The radioactive characteristics of WCCB blanket, including solid structure and functional material and the liquid water coolant, are of importance for the replacement and management of the radioactive waste. The activation analyses of the WCCB components, the primary water coolant loop, and associated operator dose rates were carried out by using the Direct Accelerated Geometry Monte Carlo Toolkit (DAGMC), and the Analytic and Laplacian Adaptive Radioactivity Analysis (ALARA) code developed by the University of Wisconsin-Madison (UW) Fusion Technology Institute (FTI). In these analyses, the three-dimensional (3-D) neutronics model was employed and the WCCB blanket were modeled in detail to provide detail spatial distribution of neutron flux and energy spectra. Then the neutron flux, energy spectra and the materials were transformed into ALARA to do the activation calculation with an assumed irradiation scenario of CFETR. The activation corrosion product (ACP) in the water coolant was also evaluated. This paper presents the main results of the activation analysis to evaluate the radioactivity, the decay heat, the contact dose, and the waste classification of the solid radioactive materials as well as the evaluate of WCCB primary coolant activation corrosion product.
Hydraulic analysis of the whole CFETR WCSB blanket module using CFD method

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A conceptual structural design of Water-Cooled-Solid-Breeder (WCSB) blanket, one of the breeding blanket candidates for China Fusion Engineering Test Reactor (CFETR), is now being carried on by Institute of Plasma Physics Chinese Academy of Sciences (ASIPP). To validate the reliability of the designed blanket module, detailed thermal-hydraulic analysis is necessary. The computational fluid dynamics (CFD) method, which can accurately simulated the three dimensional (3D) velocity and temperature fields, is a promising approach to thermal-hydraulic analysis of the whole blanket module. However, it usually takes a great deal of computing resources by 3D method due to its complicated and huge structure. Therefore, the possibility of CFD method applied for the whole blanket module needs to be deeply analyzed. In this paper, we try to study the hydraulic performance of the whole WCSB blanket module using CFD method. Firstly, each component, such as the first wall (FW), breeding region, and side wall, is simulated one by one to investigate grid independence and turbulent model. Based on these analyses the computation of the whole blanket module is finally completed. The results demonstrate that grids have a great effect on the flow distributions. Around 33 million meshes are needed for grid independence. And it takes the computer with 24 processors about 10 hours to complete one computation. The results also show that the total pressure drop of the whole blanket is close to 0.215 MPa and the flow distributions in most regions are uniform. However, the maximum unevenness of flow distributions appears in the first wall (FW). These results consequently indicate that the present design of FW is unreasonable since the mal-distribution may result in local overheating. Obviously, these 3D simulations can provide very good basis for accurate design of the whole blanket module.
Tokamak reactors like ITER or fusion DEMO reactors have serious concerns about material damages to plasma facing components (PFC) due to plasma instabilities. Plasma disruptions, such as vertical displacement events (VDE), with high heat flux can cause melting and vaporization of plasma facing materials and also burnout of coolant channels. In addition, high thermal stresses due to rapid changes of temperature can degrade the integrity of PFCs like first wall of blanket module. In order to simulate melting or vaporization of first wall in blanket module when VDE occurs, one-dimensional heat conduction equations were solved numerically with modification of the specific heat of the first wall materials using effective heat capacity method. Also, a nuclear reactor thermal-hydraulic analysis code, MARS-KS, was adopted for solving hydrodynamics in coolant channels and heat equations in other components of blanket except first wall due to its prediction capability for two-phase flow and critical heat flux (CHF) value in coolant channels. A water-cooled breeding blanket concept was selected for simulation target according to the conceptual design of the Korean fusion demonstration reactor (K-DEMO) proposed by the National Fusion Research Institute of Korea. It includes 7 mm thick first wall as plasma facing components which consists of 5 mm thick tungsten, 1 mm thick vanadium, and 1 mm thick reduced activation ferritic/martensitic (RAFM) steel. Two VDE simulation conditions were selected for first wall heat flux values of 600 MW/m$^2$ (0.1 sec) and 200 MW/m$^2$ (0.3 sec). Simulation result showed that temperatures of tungsten, vanadium, and RAFM steel exceeded each melting temperatures so that 1.55 mm thick tungsten was melted as maximum and also burnout of coolant channel occurred.
Status of the engineering activities carried out on the European DCLL

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In the framework of the EUROfusion programme, Dual Coolant Lithium Lead (DCLL) breeding blanket is being investigated as a candidate for European DEMO, which is based on the use of Pb-17Li as breeder and coolant (“self-cooled breeding zone”) and high-pressure helium for cooling the structures made of reduced-activation ferritic steel (EUROFER). During the first part of the project, a conceptual design of the DCLL equatorial module has been finalized, that meets the requirements of tritium self-sufficiency and shielding. This first design was produced to work under normal, undisturbed operational conditions; therefore no special attention was paid on accidental scenarios. Thus, a new design of the DCLL equatorial module has been produced in a further step to comply with the in-box LOCA requirement. To reach higher robustness, the number of radial stiffening plates has been increased and the walls of the main box structure have been reinforced. The DCLL helium circuit has been considerably simplified due to lower cooling needs than initially expected. A lower mass flow rate is then required and the total He pressure drop is reduced, which finally contributes to the improvement of plant efficiency. The Back Supporting Structure (BSS) consists in a unique piece which has two long PbLi poloidal ducts covering the whole segment length, having supporting and shielding functions. The rationale of the cooling scheme for the PbLi and He in the BSS is addressed. Corrosion and MHD phenomena in the BSS are also studied. This paper includes the most relevant results on neutronics, thermal-hydraulic and mechanical calculations for the present DCLL design. Advances on the Flow Channel Insert (FCI) development are also presented, with special emphasis on two different proposals: a sandwich-like FCI which uses ceramic pebble beds between two steel plates or a simple highly dense alumina ceramic tube.
Transient thermal-mechanical analysis of fusion breeding blankets: Application to the European DCLL DEMO blanket

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General purpose finite element (FE) softwares can be readily used for the stationary analysis of breeding blankets of a nuclear fusion reactor. However, the analysis of transient effects generated during the pulsed operation mode requires transient simulations to be carried out. Nowadays, a commercial tool which can be directly used for these transient simulations with affordable computational times and taking into account the coupling of the transient temperature evolution of the structure and the coolants is not available in FE commercial codes, mainly due to the very high number of cooling channels present in this type of component and the numerical stability condition associated to existing forced convective/diffusion elements available in the libraries. In the present work, a simplified but realistic methodology able to predict the evolution of the coolant temperature which can be coupled through user subroutines with the general solution of the FE representation of the structure has been developed, allowing for the resolution of the pulsed transient problem. The methodology developed has been used to analyze the feasibility of the DCLL DEMO blanket conceptual design being currently developed by CIEMAT from a structural point of view. In particular, the structural response of the blanket OB equatorial module for two representative scenarios, which are expected to drive the design, has been studied: pulsed normal operating conditions and LOCA (Loss of Coolant Accident) conditions. In order to precisely assess the structural response of the DCLL module, the corresponding capacity checks have been carried out. To that end, a global methodology which allows for the capacity check assessment of multiple failure modes (based on RCC-MR and ISDC-IC design codes) and a very large number of locations (supporting lines), combining the different time instant of pseudo-static and transient loads including gravity, pressure, thermal, seismic and electromagnetic loads has been developed.
Preliminary system modelling for the EUROfusion Dual Coolant Lithium Lead blanket

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The Dual Cooled Lithium Lead (DCLL) blanket is one of the four breeder blanket technologies under consideration within the framework of EUROfusion Consortium activities. The aim of this work is to develop a preliminary model that can track the tritium concentration along each part of the DCLL blanket and their ancillary systems at any time. Because of tritium’s nature, the phenomena of diffusion, dissociation, recombination and solubilisation have been carefully taken into account when describing the tritium behaviour inside the lead-lithium channels, the structural materials, the flow channel inserts and the helium channels. The simulations have been performed using the object oriented modelling software EcosimPro. Results have been obtained for a pulsed generation scenario for DEMO. The tritium inventory, the permeation rates and the amount of tritium extracted from the lead-lithium loop have been computed. The DCLL concept is characterised by the high velocity of the lead-lithium which cause heavy MHD effects. Because of that, the simulations include the presence of MHD boundary layers that favours the permeation along the toroidal direction over the permeation along the radial direction. Another feature of the DCLL concept is the flow channel insert. The model shows that the alumina presented in the inserts acts as a very efficient coating. As a consequence, the small gaps filled with lead-lithium flowing at low velocities become the most relevant source of permeation for this concept. Besides, a parametric study has been performed. The influence of some key parameters (PbLi mass flow, temperatures, extraction efficiencies…) has been studied independently from each other in order to analyse their influence over the whole system. The model provides valuable information for the design of the DCLL blanket. More complex upgrades are planned to be implemented based on this model in future stages of the EUROfusion Project.
P2.163

Thermal-hydraulic design of a DCLL breeding blanket for the EU DEMO

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The Dual Coolant Lead-Lithium (DCLL) is one of the breeding blanket concepts under investigation in EUROfusion. This concept is characterized by the use of self-cooled eutectic PbLi as neutron multiplier and tritium breeder and carrier, whereas supercritical helium is used to cool the first wall and other parts of the structure. The thermal-hydraulic (TH) design of the breeding blanket, as the main thermal source for power conversion, reveals itself as a key issue to counteract the influence of the foreseen low overall plant availability on the cost of electricity. In this sense, the decreased contribution of the non-breeding coolant in the DCLL presents clear advantages, like less dependence on the long-term availability of helium and lower recirculating power (recompression). The short operational range of temperature (300-550°C) imposed by the use of RAFM steel is handled by adopting the Multi-Module Segment concept. This allows lower PbLi velocities by arranging in parallel the circuits of different modules. In consequence, the magnetohydrodynamics phenomena and corrosion rates are diminished. The high Péclet numbers validate the use of simpler computational codes to couple thermally both coolants, taking advantage of assuming that the heat transfer between the structure and the fluids is one-dimensional. A TH code adapted to the transient behaviour of the pulsed operation of DEMO has been developed for sensitivity analyses. The results are compared to those obtained by FEM thermal analyses in which the PbLi and He streams are treated as fluid lines. On the other hand, the results of different CFD assessments on the performance of the helium cooling system are analyzed: the cooling of the first wall and the radial walls and the flow distribution in the general manifold. Besides, a preliminary study of the effects of the high heat generation gradient in the front poloidal PbLi channels is reported.
Liquid metal heat and mass transfer coefficients in vertical ducts with flow channel inserts

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The conceptual design of the European Dual Coolant Lead Lithium (DCLL) breeding blanket is currently being developed in the frame of EUROfusion Project. To this aim, it is of utmost interest to estimate critical flow parameters such as: (1) pressure drop and heat transfer coefficient at both helium and lithium sides, and (2) tritium permeation ratio. Pressure drop in purely hydrodynamic flows (such as in the case of helium cooling channels) has been extensively studied, and the same occurs with purely hydrodynamic heat transfer coefficients. However, in the lead lithium side, magnetohydrodynamic (MHD) effects considerably modify the flow and, thus, new correlations for pressure drop and heat transfer coefficient must be obtained. There is a large background on the estimation of MHD pressure drop, mainly focused on fully developed isothermal flows. However, there is scarce information related with heat transfer coefficient under MHD flow conditions. Here, the front poloidal duct of the DCLL breeding blanket with flow channel inserts (FCI) and helium cooling channels at the first wall is studied assuming fully developed flow. A radially varying volumetric heating is considered. A parametric study is performed for Hartmann, Reynolds and Grashof numbers as well as for FCI electrical conductivity. It is not the aim of the present study to develop new correlations, but to provide the order of magnitude of both the pressure drop and the heat transfer coefficients for lead lithium under relevant DCLL flow conditions. As a second step, and considering tritium transport as a passive scalar, i.e. without disturbing lead lithium flow, tritium mass transfer coefficient is calculated in order to study the effect of the studied flow conditions on the tritium permeation ratio.
Numerical study of laminar buoyant MHD flows under non-uniform magnetic field

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Liquid metal (LM) blanket concepts are designed by many countries due to its attractive features such as geometric adaptability, good thermal conductivity and heat carrying capacity, et al. However, they all have feasibility issues associated with magnetohydrodynamic (MHD) interactions under the environment of a strong control magnetic field and the flowing high electrical conductivity LM. The MHD effect is affecting flow distribution and stability as well as the coupled heat transfer have a profound impact on fusion reactor operation and safety. Numerical simulation is presented in this paper to investigate the influence of non-uniform magnetic field on laminar buoyant MHD flows in rectangular ducts. The code is validated by ALEX experiment data and magneto-convection phenomenon analytical solution. The MHD flowing under the different direction of magnetic field and gravity is investigated numerically. The difference of pressure distribution between heat transfer MHD flow and pure MHD flow is also compared. Keywords: MHD flows; laminar buoyant; liquid metal; non-uniform magnetic field.
Tritium and heat simultaneous non-contact recovery in vacuum

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DCLL blanket has high energy recovery efficiency. Nevertheless by several technical issues, such as MHD pressure drop, tritium permeation and energy conversion membrane corrosion, technical readiness level (TRL) of DCLL is relatively not high. To breakthrough this situation, the authors propose a new method to recover tritium and heat from liquid lithium-lead (PbLi) droplet by non-contact in vacuum. Tritium is released from droplet by advection mass transfer, and heat is emitted by radiation in a same chamber simultaneously. In vacuum, i.e. non-contact of liquid PbLi with wall and membrane, permeation and corrosion problems are expected to decrease drastically. This study aimed the temperature decrease of PbLi from 700°C to 550°C after energy recovery, followed the preceding ARIES-ST study. Tritium recovery ratio is assumed higher than 90%. Calculation was performed by dividing the recovery chamber into blocks, each falling period of 0.1 sec. The emissivity on a falling PbLi droplet is not reported yet, then lower side data, 0.25, is used to maintain a conservative stance. The heat release time showed a critical role compared with tritium release, then calculation is concentrated on the temperature transition while falling. Results showed that a droplet diameter of 0.02-0.04 mm fulfilled the requirements, and atomizing droplets in vacuum was revealed a key factor of this method. Swirl method is also examined for atomization of PbLi under high temperature. These results suggest the viability of dual non-contact extraction. However several issues such as, mutual interference of droplets, surface evaporation and assumed emissivity coefficient are neglected. Atomization of liquid PbLi is also another issue. Experimental verification is mandatory to confirm the results.
The analysis of Dual-Coolant Lead–Lithium (DCLL) blankets requires application of Computational Fluid Dynamics (CFD) methods for electrically conductive liquids in geometrically complex regions and in the presence of a strong magnetic field. Several general-purpose CFD codes allow modeling of the flow in complex geometric regions, with simultaneous conjugated heat transfer analysis in liquid and surrounding solid parts. Together with a Magneto Hydro Dynamics (MHD) capability, the general purpose CFD is applicable or modeling of DCLL blankets. This presentation describes a numerical model based on the general purpose CFD code CFX from ANSYS customized to include MHD capability using a magnetic induction approach. Numerical model involves simultaneous modelling of two different liquids in different regions of the model: helium coolant, and lead lithium eutectic. Additionally neutron heating is included in the code using three dimensional heat source distribution mapped from the results of the Attila simulations. Surface heating of the front face of the blanket is also included. Geometry of the sample blanket is introduced directly from the CAD using step file. Most of the meshing was performed automatically using CFX mesher. Special grid generation methods were used to insure accurate resolution of the near wall boundary layers including several layers of large aspect ratio prismatic elements. DCLL design also includes some narrow flow regions between SiC insert and structure. These regions were meshed using sweep method two avoid high aspect ratio tetrahedral elements. The numerical model was tested against benchmarks specifically selected for liquid metal blanket applications, such as straight rectangular duct flows with Hartmann number of up to 15000. Results for a general three dimensional case of the DCLL blanket are also included. This work is supported by US DOE Contract No. DE-AC02-09CH11466
Helium Cooling Systems for Indian LLCB TBM

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Indian Lead Lithium cooled Ceramic Breeder (LLCB) Test Blanket Module (TBM) will be tested in one half of the port no#02 of ITER. In LLCB TBM, PbLi eutectic alloy is used as multiplier, breeder, and coolant for the CB zones, and Li2TiO3 ceramic breeder (CB) is used as a tritium breeding material. The LLCB TBM consists of two helium coolant circuits, one for the TBM outer box i.e. the TBM First Wall (FW) and the top, bottom and back plates and other for the Pb-Li eutectic, which cools the Ceramic Breeder (CB) packed beds. The outer box structure is India specific Reduced Activation Ferritic Martensitic Steel (IN-RAFMS) and is cooled by high pressure-high temperature (8 MPa - 300 C) helium gas named as FWHCS. The FWHCS is to extract the incident surface heat flux from plasma and, partially, the neutronic heat deposited in the RAFMS box structure and Pb-Li interface locations. The molten Pb-Li eutectic, flows separately around the lithium ceramic breeder pebble bed compartments to extract heat from the CBs. The Pb-Li flow velocity is kept moderate enough such that its self-generated heat and the heat transferred from the ceramic breeder bed is extracted effectively. The second helium circuit extracts heat from the molten Pb-Li eutectic and is named as LLHCS. Eventually, the helium systems transfer the heat to Component Cooling Water System (CCWS) of ITER. The requisite from the two helium systems are that they shall provide the coolant at the characteristic pressure, temperature and mass flow rate required by the TBM for testing and extraction of the heat produced. This paper discusses about design parameters of the two helium cooling systems, operational states of the systems w.r.t the ITER machine operation, system’s behavior during ITER Operation, and assembly of the coolant systems in TCWS Vault annex.
P2.171

Lead-lithium facility with superconducting magnet for MHD/HT tests on liquid metal breeder blanket

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This paper gives an overview of the new facility for MHD and heat transfer (HT) tests of liquid metal breeder blanket mock-ups in high magnetic field. The facility named LIMITEF5 (LIquid Metal Tst Facility, 5 T) is under construction now in JSC “NIIEFA” (D.V. Efremov Institute). The facility includes the Lead-Lithium (LL) loop passing through the warm aperture of the superconducting magnet. Superconducting magnet is planned to be supplied in 2016 and be put in operation in 2017 with the following characteristics: - magnetic field induction is up to 5.5 T; - dimensions of the magnet “warm” working zone: diameter – 900 mm, length – 1600 mm; - winding design: low temperature superconducting split solenoid. LL loop consists of melting and feeding tanks; main loop with MHD pump, electromagnetic flow meter, calibration nozzle, heat exchanger and blanket mock-up; bypasses for LL control and purification system containing oxygen sensor, plug indicator, cold trap, LL sampler; gas-vacuum and data acquisition systems. Loop main parameters are the following: LL inventory ~ 80 liters, LL temperature – 250-350°C, LL flow rate – up to 4 m³/h, pressure developed by MHD pump – up to 0.5 MPa. LL loop will be installed in the middle of 2016 and preliminary tests will be carried out in 2016 including MHD tests in magnetic field of 1T with existing dipole magnet. The details of the lead-lithium ceramic breeder test blanket module (LLCB TBM) mock-up for MHD tests in magnetic field of ~5T which is under conceptual design are also given.
Lithium molten salts (e.g., Flibe, Flinabe) have several merits as a self-cooled tritium breeding material: low reactivity, low density and low electric conductivity. On the other hand, molten salts may cause a problem of tritium migration to the structural material of the blanket due to the low hydrogen solubility. To overcome this problem, an active control of the effective hydrogen solubility of the molten salts by mixing the powder of hydrogen-soluble metals (e.g., titanium, zirconium) has been proposed by A. Sagara, where the hydrogen can be recovered by selective heating of powders with micro-wave. An increase of 5 orders of magnitude in hydrogen solubility has been confirmed by the experiment under a static condition by J. Yagi et al. Prior to the hydraulic experiment of molten-salt/metal powder mixture in a large-scale loop, hydraulic experiment with a small-size loop using water/metal powder mixture is scheduled to investigate the effect of the powder mixing such as changes in the flow characteristics, flow behavior in the magnetic field or at bend section, erosion of the inner wall of the pipe and so on, where water at room-temperature has a viscosity comparable to that of molten salts and is suitable as a simulant of molten salts. The water loop can generate steady-state, continuous flow with a flow velocity of \( \sim 5 \text{ m/s} \). It has a part of a 90-degree bend which seamlessly connected to the downstream of the straight channel with a length enough to make a fully developed turbulent flow. At this straight channel, strong permanent magnets can be set to generate magnetic field perpendicular to the flow direction. Using this water loop, the effect of powder mixing is now under evaluation. In the presentation, initial results of this hydraulic experiment will be reported.
Progress in development and qualification of beryllium for ITER blanket first wall in Russia

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The primary reasons for the selection of beryllium as an armour material for the ITER first wall are its low Z and high gettering characteristics. For this application three beryllium grades: S-65C (USA), TGP-56FW (Russia) and CN-G01 (China) have been accepted. This selection was based on the results of the ITER Qualification Program, which included characterization and testing of material performances at transient heat loads. Russia is responsible for manufacturing of 40 % FW panels of ITER Blanket. Present FW design assumes that the FW panels will be coated with the beryllium flat tiles of 8-mm thickness. In the Russian share about 60 % of panels will be made of TGP-56FW beryllium that corresponds to ~2400 kg of Be tiles. This paper presents a progress of R&D activities on the development and improvement of manufacturing technology of the tiles made of TGP-56FW beryllium and also some actions aimed on the preparation of mass production. Necessity of the improvement has been caused by the tile dimensions change due to tightening of the requirements to the FW panels because the expected cyclic thermal loadings were reconsidered from 1 to 5 MW/m². During the preparation to mass production the line for a high purity beryllium powder fabrication has been upgraded, new pressing equipment for CIP and VHP has been mounted and commissioned, the electric discharge machine has been installed and the procedures of quality control have been revised and improved. The experimental batches of billets and tiles have been manufactured using improved technology. Additionally, this paper presents the results of comparative study on influence of transient plasma heat loads with the energy density of 0.3-1.0 MJ/m² at 250-500°C (performed in QSPA-Be facility), on the erosion and surface damage of beryllium tiles made of TGP-56FW grade and S-65C grade.
SiC-fibre reinforced tungsten-based composites for divertor

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Tungsten-based composites have gained considerable attention owing to their excellent performance levels at high temperatures due to exceptional high temperature properties such as a high melting point, good thermal conductivity and a low thermal expansion coefficient. However, tungsten is also associated with a serious reduction in its strength at elevated temperatures, which is also one of the main drawbacks of tungsten for being used as a plasma facing material in fusion reactors. The main objective of this research is to develop a technique for fabrication of long fibre-reinforced tungsten, with the aim to overcome the main drawback of the tungsten as a material for divertor. The materials were prepared from tungsten powder with a small addition of TiH2 powder, a Cp-Ti foil and C-coated SiC fibres with 100 µm diameter. In this stage, the fraction of the fibres was kept small (<10 %) and the C-coating was used to prevent/minimize the unwanted chemical reaction between SiC fibres and W matrix. The samples (150 mm x 100 mm) were prepared by laying-up the material into tooling using three different sequences and compacted with a hot isostatic press. The characterisation of the prepared samples comprised of macro- and microstructural analysis, X-ray diffraction analysis and measurements of mechanical properties (bend strength and hardness). We have observed that the fibres appeared to be distributed unevenly in the composites which did not contain the Cp-Ti foil. It was also seen that the fibre reinforcement causes increased porosity and a decrease in the composites density when compared to the matrix composed of only W powder.
P2.175

Tungsten carbide particles-reinforced tungsten for divertor

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The main aim of the work has been to improve properties of the plasma-facing material for the divertor to resist high thermal loading during operation. Among the available materials we selected (carbide) particles reinforcement of tungsten, wherein the reinforcement should not chemically react with the matrix. In this respect, W2C particles offer the most attractive solution. The paper will present two fabrication routes for W-W2C composites with the composition involving up to 10 vol. % W2C precursor. As a carbon precursor, graphene or phenolformaldehyde resin was mixed with the tungsten powder. Mixtures were dry pressed and sintered in vacuum or in hydrogen atmosphere. The samples were characterized with respect of microstructure, where the main efforts were put on the identification of the newly formed W2C, and composites’ phase composition, charcaterized by XRD -Rietveld analysis. It has been confirmed, that two-phase composites (W-W2C) can be prepared by both proposed techniques. As expected, the amount of W2C particles in W-matrix increases with increasing addition of the precursor. At the same time also the density is increasing from 87 % of theoretical density for the pure W to 96 % of theoretical density for the composites in which 10 vol. % of carbon precursors were added. Microstructural analyses revealed that the W2C particles are mostly located at the W-grain boundaries and hence they successfully prevent grain-growth that occurs during the sintering. The introduction of reinforcements into the W-matrix has not only changed the microstructure, but also the mechanical properties, such as increased hardnes of the composites. In continuation, high heat flux tests will be performed to estimate the behaviour of prepared composites in order to determine whether these composites are appropriate candidates for a plasma facing material in divertor.
“FAST brazing” technology for W multi-metal laminates

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W has the highest melting point of all metals, good high temperature strength, high creep resistance and a high thermal conductivity. These properties make W a first choice for armor materials in fusion energy reactors. Unfortunately W can not be also used for structural applications, due especially to its high temperature brittle- to-ductile transition (DBT). However, when cold rolled at about 400°C, W foils, as opposed to bulk W show exceptional properties in ductility, toughness and DBT. Attempts to transfer these properties from W foils to W-based bulk materials resulted in the so-called “W-laminates” concept, i.e. multi layered composites from alternate W and other metal foils. Different approaches were already successfully used to create such composites, but several shortcomings were observed at high temperature exposure or during neutron irradiation. Here we present microstructural and thermo-physical properties results obtained for W-multi-metal laminates, using Cu, Cr or Au deposited thin interface layers and V or Ti foils. The W-laminates have been produced by FAST (field assisted sintering technique). The obvious advantage of this route resides in the short processing time, with lower recrystallization detrimental effects, while allowing for temperatures close to the metal melting points, thus making the process similar to brazing. The deposited layers can tune the Joule heating at the interfaces during processing time and consequently improving the interface microstructure. More important the multi-metal approach creates a promising route to improve the performance of materials during high temperature exposure.
Self-passivating tungsten alloys of the system W-Cr-Y: characterization and testing

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Tungsten is presently the main candidate material for the first wall armour of future fusion reactors. However, if a loss of coolant accident with simultaneous air ingress into the vacuum vessel occurs, the temperature of the in-vessel components would exceed 1000 °C, leading to the undesirable formation of volatile and radioactive tungsten oxides. A way to prevent this serious safety issue is the addition of oxide-forming alloying elements to pure tungsten which, in presence of oxygen at high temperatures, promote the development of a self-passivating oxide layer and protects tungsten against further oxidation. In this work, bulk tungsten alloys of the W-Cr-Y system with different concentrations of the alloying elements are studied in order to establish their optimum composition for lowest possible oxidation rate and best self-passivating behaviour together with acceptable thermal and mechanical properties. The materials are manufactured by mechanical alloying and subsequent densification by hot isostatic pressing. Microstructural investigations of the bulk material and the thin oxide layer developed after oxidation as well as the thermal conductivity and mechanical properties of the alloys at different temperatures are presented. The W-Cr-Y alloys exhibit an ultrafine grained microstructure with an average grain size around 100 nm. A summary of the results of different tests under conditions relevant to the expected operation as first wall material is shown: oxidation tests under isothermal and accident-like conditions; high heat flux tests at GLADIS (Garching Large Divertor Sample Test Facility) up to 2 MW/m², according to the power load expected at the blanket first wall; and thermal-shock tests at JUDITH (Juelich Divertor Test Facility Hot Cells) to simulate e.g. loads by photon flashes occurring at the first wall of a DEMO. Compared to previous alloys of the system W-Cr-Ti, the W-Cr-Y alloys exhibit significantly lower oxidation rates both under isothermal and accident-like conditions.
The study of irradiation defects structure in tungsten: stability for the self-interstitial

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Irradiation damage research is one of the basic issues to solve the application of first-wall materials in fusion engineering. The diffusion and recovery of the defects can greatly affect the performance of the materials in fusion. The rotation, stability, migration of the self-interstitial atoms (SIAs) in defect structures of tungsten is investigated by the first-principle method. It is found that the \(<11h>\) dumbbell have a lower formation energy than \(<111>\) dumbbell and a higher local charge density distribution in the self-interstitial atoms rotation. Further confirmation has been done from the uniaxial strain that the uniaxial modulus M of \(<11h>\) has a minimum value of 518Gpa in crystal tungsten. The results give a powerful evidence for the microscopic mechanisms of the stability of \(<11h>\) SIA style. Nevertheless, the \(<11h>\) SIA configure played an important role in the recovery process in irradiation damage.
Structural and micromechanical investigation of multi compositional tungsten thin film alloys produced by magnetron sputtered co-deposition process

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In order to investigate possible enhancement of mechanical properties of tungsten (W) based materials by solid solutions and to examine the influence of a single alloying element on a particular property such as ductility, a versatile production method of generating a wide range of different tungsten binary alloys is presented. Magnetron sputter co-deposition was used to produce thin films of W binary alloys with defined compositional gradients, on 200 μm thick cold-rolled tungsten substrate, yielding film thicknesses between 1-2,5 μm. Three alloys were studied: W – Fe (0-6at %), W – Ti (0-12at %) and W – Ir (0-10at %). Using this combinatorial materials method, detailed microstructural and micromechanical investigations can be performed.

A scanning electron microscope equipped with electron backscattered diffraction detector was used to determine the microstructure of the alloys, analyse the texture and the grain orientation, and to investigate the variation of the microstructure in respect to the alloying content. Different heat treatments reveal a significant grain growth above 1300°C with a clear texture transition when reaching higher temperatures. Mapping of chemical composition as a function of position on the wafer / tungsten substrate was performed using energy dispersive X-ray spectroscopy and the surface roughness was obtained with atom force microscopy. Micromechanical testing techniques, including 3-point bending experiments and nanoindentation, were performed in order to study fracture behaviour and ductility of produced tungsten film alloys.
Cu-based composites as thermal barrier materials in DEMO divertor components

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For DEMO fusion reactor an expected heat flux of about 10 MW/m$^2$ should be extracted by the divertor which will have, most likely, an armour part made of W and a following heat sink part made of Cu or ODS Cu alloy. Unfortunately, for these materials the optimum operating temperature windows do not overlap. Thermal barrier materials are interface materials included in such components, aiming to keep the temperatures of both armour and heat sink parts in the corresponding operating windows on one side, and to mitigate the effects of their different thermomechanical properties, on the other side. Here we propose a simple spark plasma sintering route to create Cu-based composites with a high content (10–40 volume %) of various dispersed materials (Al or Y oxides, C, SiC or W), allowing a fine tuning of the content and a large pool of predefined shapes and dimensions. The resulting specimens can be further joined to armour and heatsink components via a similar electrical field assisted technology. Microstructural and thermal properties are investigated for these materials allowing to select the most suited materials in view of their thermal conductivity and thermal expansion coefficients.
P2.182

Thermal conductivity and diffusivity of Cu-Y alloys produced by different powder metallurgy routes

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Copper-based materials are considered the most promising candidates for water-cooled components of the heat sink systems of future fusion reactors. Although pure copper is the material with the higher thermal conductivity, the detriment of its mechanical strength on increasing temperature restricts its use at high temperature. In the last years, ODS Cu-Y2O3 and Cu-Y alloys have been produced following PM (powder metallurgy) routes and subsequent consolidation by HIP (hot isostatic pressing). The different approaches include the milling of pure copper with nano-sized particles of Y2O3 or pure Y powders, and the utilization of prealloyed Cu-0.8%Y powders obtained from vacuum induction melting and atomization. Microstructural studies showed that Y and/or Y2O3 particles are formed and are homogeneously distributed in the copper matrix. Tensile tests performed in the temperature range 293-773 K indicated that the presence of these particles give place to a reinforcement of the matrix copper. The effect of the equal channel angular pressing (ECAP) thermo-mechanical treatment in the microstructure and mechanical properties was also evaluated. In this work, the thermal properties in the temperature range 300 - 773 K of ECAPed and non-ECAPed Cu-1%Y2O3 and Cu-0.8%Y alloys will be presented. The thermal expansion has been measured by using a dilatometer. The thermal diffusivity and specific heat have been determined by the Laser Flash technique. From these measurements, the thermal conductivity of the alloys has been estimated. The results indicate that the ECAP process enhances the thermal conductivity of the alloys.
P2.183

Application of Friction Stir Processing for Mechanically Strengthening Pure-Cu and CuCrZr alloy

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Copper is the candidate material for cooling components for divertor and other plasma facing components. Although CuCrZr alloy is a first choice regarding strength, toughness, and conductivities, issues related to quality control during manufacturing process and also on the possible loss of strength during brazing among fabrication of the components still remains. CuCrZr also exhibit some weakness against neutron irradiations, e.g. pronounce loss of ductility at very low doses below 150˚C. One of the keys to deal with these issues is a grain refinement. Friction Stir Processing (FSP) is a solid-state process in which a rotational tool is plunged into the work piece to produce local friction heating inducing complex material flow and intense plastic deformation. The important feature of this process is strengthening but also is ultra-grain refinement, which is expected to improve irradiation resistivity. The purpose of this study is to examine the applicability of FSP to copper and its alloys to improve material’s performances. In FSP, material flow induced by friction heating depends on the rotation speed. To examine the effect of rotation speed, variety of rotation speed from 50 to 500rpm was tested on pure Cu with fixed vertical force of 1.5t. The result indicates that the most effective grain refinement can be achieved at the rotation speed of 100 to 200rpm with hardness increase of about 50%, but the introduction of the defects like cavities and local cracks were also seen. For better FSP performance, compulsory cooling with liquid CO2 during the process was also tested and finer grain size was achieved compared to non-cooling tests at the same and greater rotation speeds, and consequently defect formations was also suppressed. Based on these results on pure Cu, the examination on ITER-Gr CuCrZr alloy was also conducted. The details will be reported on the presentation.
Development of Oxide dispersion strengthened-Copper using MA-HIP process

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Copper (Cu) alloy is a candidate materials for use as heat sink materials of fusion divertor because of its good thermal conductivity. In recent years a number of studies have been carried out on Cu-based materials such as Precipitation Strengthened Cu (PS-Cu). However, the material has some critical issues such as instability of microstructure at high temperature and loss of strength by irradiation induced softening and hardening. On the other hands, conventional dispersion strength (DS)-Cu such as GlidCop\textsuperscript{\textregistered} (Cu-Al\textsubscript{2}O\textsubscript{3}) is known to have higher stability in microstructure at high temperature than PS-Cu. These conventional DS-Cu, which have been produced by internal oxidation and extrusion, may cause coarsening of the dispersed particles, inhomogeneity in microstructure and anisotropy in mechanical properties. In this study, a new DS-Cu alloy was fabricated by combination Mechanical Alloying (MA) and Hot Isostatic Pressing (HIP) method. This alloying-sintering technique is well known as a technique which can enhance mechanical properties at high temperature. A good example is seen in Oxide Dispersion Strengthened Steel. Therefore, we expect that MA-HIP process can offer technical advantages for DS-Cu. Examination of the particles after MA shows that the grain size and Vickers hardness decreased and increased, respectively with the increase in MA time. The mechanical alloyed copper from 8 hr to 32hr exhibited different electrical resistivity. This results suggest that the microstructure and the strengthening mechanism have changed between 8 and 16 hr of the MA process. At MA time of 32 hrs, the hardness was comparable to that of Glidcop\textsuperscript{\textregistered} although the grain size is much larger. In result of the texture observation using TEM, equiaxed-grains which contained nano-particles with high Al levels were observed. These results implied that the precipitate hardening took place for the alloy more significantly than that for Glidcop\textsuperscript{\textregistered}. 
Characterization of HIP joints of a simplified prototype of ITER NHF First Wall Panel

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The blanket is one of the most critical component of ITER. It is directly exposed to the plasma and acts as shielding of the vacuum vessel from the neutrons and other energetic particles produced in the fusion plasma. Each of the 215 Normal Heat Flux (NHF) panels consists of a shield block and a First Wall (FW) panel. The NHF FW panels consist of a complex bimetallic structure of 316L stainless steel (SS) backing plate and a copper alloy (CuCrZr) heat sink, covered with beryllium armor tiles. Joining of these materials is done by HIP, following the solid state diffusion bonding technique. Under the framework of a R&D roadmap parallel to the manufacturing of the full-scale prototypes of a FW panel of ITER, this work shows the characterization of the joints involving AISI-316L and CuCrZr produced by diffusion bonding by HIP of a simplified 10-fingers prototype of a FW panel. This 10-fingers prototype has been developed by Leading Metal Mechanic Solutions to design proper cutting strategies of the fingers of the FW panels of ITER to mitigate distortions in the final component. Microstructural and mechanical characterization of the raw material and the joints AISI-316L/AISI-316L, AISI-316L/CuCrZr and CuCrZr/CuCrZr is presented. Mechanical tests were performed following ITER recommendations and the results were compared to F4E specifications. Exhaustive microstructural characterization of the interface was performed by Scanning Electron microscopy (SEM), including Energy Dispersive X-ray Spectroscopy (EDS) and Electron Back Scattered Diffraction (EBSD). The strength of the joints was correlated with the quality of the surface preparation before bonding and the presence of foreign oxides at the interface. The phases developed at the interface during HIPping and post-HIP heat treatments were analyzed. Finally, the effect of the manufacturing thermal history of CuCrZr on its microstructural evolution and its mechanical behavior was investigated.
The actuality of the topic comes from the ITER (International Thermonuclear Experimental Reactor) fusion tokamak that is a major international experiment with the aim of demonstrating the scientific and technical feasibility of fusion as an energy source. Among others the most challenging task is to find proper materials and technology for Plasma Facing Components. Welding by HIP (Hot Isostatic Pressure) or diffusion bonding methods as a candidate solution between Plasma Facing Components generated significant technological investigations over the last decades. There are several areas where it can be adapted: the shaping of cooling channels inside the Blanket Modules by grooves require diffusion welding of large surfaces, while the welding of plasma facing tiles as Be and W to heat sink materials is foreseen at first wall and to Divertor cooling channels. These require HIP, or similar diffusion methods as HRP (Hot Radial Pressure welding). These technologies can result in mechanical degradation and microstructural changes at the bonding surfaces. As a consequence, microstructural characterization is required for standardization of the HIP technologies. This paper intends to summarize the following topics in the diffusion bonding subjects: the use of metallic interlayers between welded materials for compensation of their different physical properties; the formation and effect of intermetallic layers in the transition area; the impact of thermal fatigue at welded surfaces; the behaviour of ODS steel in the in the conditions of HIP welding technologies. This paper is going to summarize the state of the art of high standard knowledge of the four technical fields of Plasma Facing Components: using of interlayers, the effect of intermetallic layers, impact of thermal fatigue and the behaviour of the ODS steel during HIP welding process.
Study on the optimization of Cu/SS explosion bonding for ITER first wall components

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The CuCrZr/316L(N) explosion bonding bimetallic plates were used to make hypervapotron (HVT) cooling channel for the fingers, which is the key components of the ITER First Wall (FW). The bimetallic plates will be subjected to the same thermal cycles as the FW component, including the HIP (hot iso-static pressing) joining for bonding HVT and beryllium tiles, thus the properties of both the CuCrZr/316L(N) joints and the CuCrZr alloy will be changed even cannot meet the design requirement. Several batches of CuCrZr/316l(N)-IG joints were obtained by explosion bonding. Visual test and ultrasonic examination were conducted for the explosion bonding plates, samples were cut from the plate in different conditions for the tensile testing, metallographic examination (microscopic and macroscopic) and hardness testing. The results shows that a solid solution annealing heat treatment was necessary to recover the tensile strength of CuCrZr alloy as well as the deformation of CuCrZr/316L(N) joint interface so that the risk of stress inducing corrosion will be significantly reduced. In addition, the optimization of manufacturing process for the CuCrZr/316L(N) bimetallic plates and the HIP parameter will benefit the properties control of the materials.
Development of new materials is one of the key for the construction of the new fusion power plant (DEMO). The selected materials have to fulfill several requirements such as standing the conditions that takes place in the core (high neutron flux and temperatures close to 1200 °C) and low activation rate. Several techniques have been proposed to join the different parts of the first wall components such as brazing, diffusion bonding and laser welding. Among them, high temperature brazing seems to be the most suitable technique due to its limited effect on the base material properties. Cu-20Ti filler composition has been demonstrated to fulfill all the joint requirements to be used in the first wall of fusion reactors [1]. The filler is made of a mixture of pure Cu and Ti powders. However, the usage of alloyed powders instead of pure powders provides some advantages. Results of W-EUROFER joints using Cu-20Ti alloyed powders have shown the consecution of high quality joints. The filler melts in a reduced time compared to previous one because the as-received powders already have the Cu-20Ti composition. This fact enhances the spreading capabilities of the filler at the joint clearance and opens the possibility of developing the joint with lower brazing temperatures. Moreover, the use of alloyed powder allows working with higher heating rates (up to 6 time higher), saving hours of heating process. The improvement could benefit both for reducing diffusion effects in EUROFER and improving productivity of the brazing procedure. The characterization of the joint has been done by means of SEM and optic microscope for the microstructural characterization and shear strength and Vickers microharness tests for mechanical characterization. [1] J. de Prado et al. Development of brazing process for W– EUROFER joints using Cu-based fillers. Phys. Scr. T167 (2016) 014022 (5pp).
Radiation tolerant optical components of future fusion reactors have to withstand radiation of unprecedented intensity. It is widely recognized that spinel lattice of AB2O4 double oxides demonstrates enhanced resistance against neutron irradiation. Therefore, the development of spinel optical materials and understanding of their radiation damage processes is of great importance. One defect type of spinel lattice, the so-called antisite defects, needs special attention because these defects influence both, the sintering process and optical properties of resulting ceramics [1]. Antisite formation in AB2O4 spinels means swapping the positions of A and B cations between tetrahedral and octahedral interstices. The deviation from the perfect (normal) spinel structure caused by this swapping is characterized by the inversion parameter i, defined as \((A_{1-i}B_i)[A_iB_{2-i}]O_4\), where the parentheses refer to the tetrahedral oxygen coordination and square brackets to the octahedral one. The change in i of \(\sim 0.1\) modifies the lattice parameter by the value that can be determined by XRD, while optical properties are influenced by very low i values, for example, in the case of MgAl2O4 spinel the inversion of \(i=10^{-3}\) corresponds to antisite concentration of \(1.5\times10^{19}\) cm\(^{-3}\). It is a huge amount in terms of material doping, and these antisite defects cause considerable changes in optical properties. In this study, MgAl2O4 optical ceramics and single crystals irradiated with protons or fast neutrons have been investigated. Besides optical absorption, highly sensitive luminescence methods (cathodo- and thermally stimulated luminescence) have been used to reveal the manifestations of antisite defects and to elaborate the criteria for the determination of a low-level inversion in MgAl2O4. [1] N. Mironova-Ulmane et al, Rad. Meas. in press.
First mirror (FM) lifetime is one of critical issues for the optical diagnostic system in ITER since it greatly influences the performance of relative diagnostic. In ITER, repetitive cleaning is expected to give a positive solution to the frequent replacement of FM, thus prolonging its lifetime. Three cleaning cycles using radio frequency argon plasma were applied to the stainless steel mirror with the dimensions of 35mm×30mm×2mm to understand the evolution of mirror optical and morphology performance during each cycle. The aluminum film with the thickness of about 100nm was deposited on the mirror surface under the identical conditions in each cycle. When three cycles were implemented with same cleaning parameters, it was found that with the complete removal of film, the total reflectivity was restored up to 98% of its initial value. Nevertheless, the roughness grew as more second phases and the deformation of grain boundaries arose, leading to increase of the diffuse reflectivity of 16% and 27% in the second and third cycles, respectively. It was explained by excessive sputtering due to the formation of thinner film and lower adhesion strength between substrate and film which were measured by Ellipsometer and Nanoindenter, respectively, on the rougher surface. After lowering the selfbias during the last two cycles and with shorter cleaning time in the third cycle, the diffuse reflectivity was improved to about 10% and 15% on the basis of the good recovery of total reflectivity, respectively. In conclusion, at given conditions every repetitive cleaning cycle will increase the surface roughness, accordingly degrading the specular reflectivity, and inevitably have a negative effect on FM lifetime. It is highly essential to improve the mirror specular reflectivity by optimizing the cleaning parameters in each cycle in order to maintain its specular reflectivity as high as possible.
P2.192

The consequences of fundamental design choices for DEMO

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There are a number of key design difficulties in producing an integrated demonstration fusion power plant (DEMO) design, and how these issues are resolved fundamentally affects the final overall design. Technological examples include the issue of power loading in the divertor and reducing recirculating power through efficient current drive. Additional drivers include economic considerations such as acceptable capital cost; the target build date of the plant; and whether or not the plant is intended principally as a technology demonstrator for fusion or is a ‘nearly-commercial’ power plant, which can be moved into production with only minor further developments of physics or technology. This contribution explores how the varied approaches to solving these often-competing demands results in very different demonstration power plant concepts from different groups globally, even ones using similar physics and technology models. One of the most important considerations in this analysis is the target build date: a near-term DEMO must use known and well-characterised regimes of operation and technology with a clear development path, rather than speculative concepts and scenarios which might only be attempted in ITER late in its operational phase. This in turn makes the jump from DEMO to a commercial power plant, which must be economically competitive and have high availability, much larger and so places higher demands on DEMO as a development facility. The gap from DEMO to a power plant is also considered.
Breeding blanket research and development is recognized as one of the most important areas for realizing an energy-producing fusion reactor. In China, the ceramic breeder/helium coolant/ferritic steel structure is considered as the main concepts of the blanket to conduct the breeding blanket research, and on the other hand, the liquid breeder blanket is also to be investigated as the alternative option. Helium cooled ceramic breeder (HCCB) TBM is one of China’s TBM concepts. HCCB TBM uses Li4SiO4 pebble as tritium breeder and Beryllium pebbles as neutron multiplier. The structure material is reduced activation ferritic/martensitic steel. High pressure (8 MPa) helium will flow through the cooling channels inside structures to carry out heat. The generated tritium will be extracted by low pressure (about 1 atm) helium purge gas flow through pebbles. The dual functional lithium-lead test blanket module (DFLL TBM) is another concept of TBM concepts. DFLL TBM design has the flexibility of testing both the helium-cooled quasi-static lithium-lead (SLL) blanket concept and the He/PbLi dual-cooled lithium-lead (DFLL) blanket concept. In this paper, the RAMI approach is used to compare those two TBM conceptual designs. The two conceptual designs have been compared from the function, risk level, system reliability and operation availability. In the compare progress, the function breakdown was performed on those two conceptual designs. Based on the result of function breakdown, the reliability model were established for those two TBMs and the inherent availability are 94.69% for HCCB TBS and 98.57% for DFLL TBS over two years based on the ITER reliability database. Besides, the Failure Modes Effects and Criticality Analysis (FMECA) is also performed on those two TBMs with criticality charts highlighting the risk level of the different failure modes with regard to their probability of occurrence and their effects on the availability.
Time-dependent power requirements for pulsed fusion reactors

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The investigation of time-dependent power requirements for a future nuclear fusion reactor is part of the systems integration task for the European Fusion Programme. All fusion power plants, whether pulsed or steady-state, will require electrical power to operate the various plant systems. Over the entire pulse cycle reactor systems will require varying levels of power over different time periods. For example, the heating and current drive system will require power during start-up, burn and ramp down phases. A number of these systems will require power continuously and outside the pulse window, such as the vacuum systems or the superconducting toroidal field (TF) coils. This paper presents the recently updated modelling of these power requirements over the whole pulse cycle for all plant systems in the systems code PROCESS. For each system (and subsystem) the operational states are defined and the power required during each state estimated. Combining the systems and subsystems provides the overall energy usage at different times of the pulse cycle. The power requirements are summarised and the overall power generated versus power required is output. Having the systems code include this modelling in its optimisation is essential for determining the reactor power balance and overall viability of the fusion power plant.
P2.195

Dynamic modelling of a water-cooled blanket and energy storage options for a pulsed DEMO

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The Water-cooled Lithium-Lead (WCLL) blanket is one option under consideration for the EUROfusion DEMO programme. This blanket design must interface with the Primary Heat Transfer System, Power Conversion System, and Energy Storage System in an integrated solution to mitigate the pulsed power profile of the tokamak and deliver feasible power plant performance. The system must maintain an acceptable electrical output during the dwell period and minimise thermal and mechanical cycling of components and systems. This work presents the development of a complete transient model, constructed using the Apros simulation code, of the WCLL blanket with heat transfer and thermal hydraulic phenomena described in one dimension. The model extends to include all primary coolant loops, the LiPb circulation loop, and the secondary steam cycle, with all appropriate components and control systems. Two energy storage options are investigated: first, use of the LiPb breeder material itself as a latent heat energy storage medium that continues to deliver heat to the coolant via the blanket modules during the dwell; and second, to reduce the volumes of LiPb necessary, the introduction of an interfacing molten salt storage loop. For these options the model captures the thermal inertia of the system, providing the time-dependent plant response. Specific topics of interest include temperature and pressure transients in the various fluid loops, volume requirements for the storage media, and ultimately the power profile delivered to the grid. 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 and from the RCUK Energy Programme [grant number EP/I501045]. To obtain further information on the data and models underlying this paper please contact PublicationsManager@ccfe.ac.uk. The views and opinions expressed herein do not necessarily reflect those of the European Commission.
P2.196

Analysis of the secondary circuit of the DEMO fusion power plant using GateCycle

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ITER is planned to be the research type tokamak which will achieve the energy breakeven point. The next step towards the realization of fusion energy will be DEMO – the first demonstration fusion power plant producing grid electricity at the level of a few hundred MW. DEMO designers are required to maximize the conversion efficiency of the primary and secondary plant circuits. The Primary Heat Transfer System (PHTS) transfers heat from the nuclear heat sources, i.e. blanket, divertor and vacuum vessel, to the secondary circuit called Balance of Plant (BoP) responsible for generating electric energy. Two main candidate options for the realization of PHTS are considered, namely: Water-Cooled Lithium-Lead Breeding Blanket (WCLL) and Helium-Cooled Lithium-Lead Breeding Blanket (HCLL). The present work is focused on modelling the candidate steam/water BoP cycle for the HCLL option using the GateCycle software. Operation at nominal conditions (during the plasma burn) and in the “off design” conditions (for the dwell phase) is analysed. Since we utilize in the simulations the model of the turbine which has not been produced yet, its part-load characteristics are determined according to the methodology proposed in GateCycle. The developed model of the BoP cycle is used to study the impact of the operating conditions on the cycle efficiency and its basic operating parameters.
The cooling system is one of the key parts of the fusion power reactor technology. The DEMO fusion power reactor should have different heat sources (first wall, blanket, and divertor) with different temperature and power. In the current European concept of DEMO, helium and water are used as the cooling medium. However, use of Helium and water introduces some issues in terms of their properties and also in the design of the systems. A possible alternative is the supercritical carbon dioxide (S-CO2) cooling system. This paper focuses on the application of S-CO2 for the fusion power reactor. In the first part the design of cooling system with S-CO2 for primary circuit is completed and compared with the existing options. Then the design of the S-CO2 loop for secondary circuit is researched with different medium on the primary side (water, helium and S-CO2). Last part of this paper is focused on the study of the S-CO2 thermal cycle and efficiency estimation. The reason for research of the cooling systems with S-CO2 for the fusion power reactor is because the compression work of S-CO2 is lower than with He and also because S-CO2 cycle components are more compact than both steam Rankine cycle and helium Brayton cycle. The heat addition into a thermal cycle from different heat sources with different temperatures is generally disadvantageous in terms of efficiency of Rankine-Clausius and helium cycle. However, this is actually advantageous for the S-CO2 cycle as it resolves the real S-CO2 property issues. S-CO2 is thus an attractive option for heat removal as well as conversion of heat into energy and might have overall better performance than steam Rankine-Clausius or helium Brayton cycles. Therefore, it should be considered as an option for the DEMO fusion power reactor.
P2.198

Preliminary safety analysis of LOCAs in one EU DEMO HCPB blanket module

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HCPB (helium cooled pebble bed) blanket concept is one of the EU DEMO blanket concepts running for the final design selection. It is necessary to study the pressure behaviour in the blanket and the connected systems during the loss of coolant (LOCA) in a blanket module, as well as the temperature evolution in the coolant flow and the associated structures. The LOCA can be caused by rupture/leak of sealing weld or cooling channels inside the blanket box. Concerning cooling channel locations in the HCPB blanket design (version 2014), which are identified as the first wall (FW), the horizontal and vertical plates (HP, VP) of the stiffening grids (SG), and the cooling plate (CP) of the breeder unit (BU), three representative accidental sequences for the design basis accident (DBA) have been assessed: case I for a failure of one HP in the SG, case II for a failure of 10 channels in the FW, and case III for a failure of one CP in the BU. Case I and III are called in-box LOCA, while case II is in-vessel LOCA. In case I and II helium ingress into the vacuum vessel (VV) will enter into the expansion volume (EV) by exceeding the VV pressure limit. In case III helium ingresses into the purge gas system, this connects the EV for pressure relief. MELCOR 1.8.6 for fusion is used for the analysis of LOCAs. Two separate loops are modelled for the redundancy of the primary heat transport system (PHTS). The accident is initialised during the normal operation at the steady state. Pump is shut down in 3 s after the LOCA, while a fast plasma shutdown (FPSS) is activated in 4 s. The transient results are presented in this paper and the impact of the FW break size is discussed as well.
Radioactive toxins confinement is a main safety function for nuclear power plants, hence the importance of confinement design parameters optimization. In this context, performing parametric assessments of thermodynamic variables thought to be relevant for confinement design can help at better framing the option design space. In the context of DEMO EUROfusion WP, FFMEA studies are going on for the selection of the most significant accident sequences to be analysed in a deterministic way. The FFMEA results are not available currently. In the meantime some sensitive analyses have been done focusing the attention on few expected relevant accidents as selected loss of coolant. Moving from this perspective, present work focuses on two LOCA accidents relevant to Vacuum Vessel Primary Heat Transfer System water cooling loop and to Toroidal Field helium cooled coils. In particular the consequences of cooling inventory loss into gallery rooms surrounding the tokamak are explored. The importance of such accident investigation resides in the fact that these two LOCA accidents may impact galleries design parameters as they constitute a second confinement boundary for radioactive toxins. Building confinement is commonly required to be less performing with respect higher level barriers, so that this explorative analysis may support the definition of such performance requirements and also might suggest corrective actions to mitigate the accident. Based on publicly available ITER data, a first approximation scaling to DEMO is obtained and a set of sensitivity simulation analyses are performed on main variables (coolant inventories, enthalpy, rooms volume, etc.) in order to derive resulting galleries pressure and temperature conditions. A first feedback to the design of the systems considered is reported too. The deterministic analyses are simulated by mean of CONSEN5, a fast running thermohydraulics code, which relying on simplified input models, is suitable for such explorative analyses.
Effective water cooling of very hot surfaces during the LOCA accident

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The first wall, blanket and divertor targets provide a physical boundary for the plasma influence and have to be intensively cooled during the operation in case of the high power fusion reactor. In the case of the LOCA accident, the released fusion power can be stopped very quickly, but the final plasma disruption may load the non-cooled components, and a large amount of heat accumulated in the component material needs an emergency cooling. Moreover, the decay heat power in the range of a few percent of the fusion power can destroy components, if not emergency cooled. Heat transfer during cooling of very hot surfaces with subcooled liquid is still not fully explored area. When water meets a hot surface, a thin steam layer is presented between the coolant and cooled surface. Heat removal can be successful only when water rewets the surface. The place where water rewets the surface is called the quench front. In the paper, the quench front propagation along the cooled geometry highly influenced by initial wall temperature, coolant flow rate and by heat accumulated in the cooled components. Understanding the phenomenon of rewetting of hot surfaces is crucial for reactor safety. The study is focused on the quench front propagation in the annular channel with initial wall temperatures within range 250 – 800°C and coolant flow rates from 100 kg/m²s up to 1200 kg/m²s. Also, the influence of heat capacity of the cooled wall is investigated. For this purpose, an experimental loop with a variable hydraulic circuit including the test section with a length of 1.7 m has been built, and results of the experimental research are presented.
P2.201

Qualification of MELCOR and RELAP5 nodalization models for EU HCPB TBS accident analyses

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‘Fusion for Energy’ (F4E) is designing, developing, and implementing the European Helium-Cooled Lithium-Lead (HCLL) and Helium-Cooled Pebble-Bed (HCPB) Test Blanket Systems (TBSs) for ITER. Safety demonstration is an essential element for the integration of these TBSs into ITER and accident analysis is one of its critical components. The F4E, Amec Foster Wheeler and INL comprehensive methodology for fusion breeding blanket accident analysis, published last year, consists of several phases. The methodology starts with the selection of reference accident scenarios, the development of detailed accident analysis specifications and the assessment of analysis codes. Models of the TBS are then constructed using the selected codes (MELCOR 1.8.2 and RELAP5-3D for the HCPB TBS) and modelling approaches. The models are qualified according to a test matrix including comparison with TBM finite element design analyses, code-to-code comparisons (between the MELCOR 1.8.2 and RELAP5-3D models) for both TBS normal operation and transient cases, and sensitivity studies for accident scenarios. The qualification test cases that are executed gradually move from models of separate systems to a complete TBS model, and from the simulation of steady-state and normal plasma pulse operation to consideration of power excursions, operational transients and accident events. Finally, both of the qualified models are used to analyse a selected accident scenario (a 32 hour loss-of-offsite power) together with sensitivity studies dedicated to the evaluation of uncertainties. This step completes the qualification process. The impact of uncertainties associated with the accident analyses is also addressed to provide sufficient confidence in the level of conservatism in the results. Following an expert review of areas of uncertainty (including phenomena identification and ranking table (PIRT)) a gradual approach to uncertainty assessment has been adapted. The results obtained in the qualification of the EU HCPB TBS models and their uncertainty evaluation will be reported in the paper.
Simulation experiment on pressure shock waves under blanket in-box LOCA

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The helium cooled LiPb blanket concept has become a promising design for fusion reactors in the world. Considering the complex design of the blanket, it is likely that helium gas leakage into the liquid alloy may occur due to tube rupture, named in-box Loss of Coolant Accident (in-box LOCA). And corresponding shock waves likely occurred at the break position and transferred within the liquid metal, causing the sudden pressure jump into LiPb metal. It will raise the great challenge to the integrity of the blanket. A Chinese multi-functional liquid metal experimental platform safety test loop (KYLIN-II S) was constructed to investigate the interaction between gas and LiPb for in-box LOCA. The facility consists of a reaction vessel, gas tank and a safety tank. During the gas injection into the reaction vessel, the high precision piezometric pressure transducers placed on the vessel wall were used to measure dynamic pressure in gas and liquid metal. The preliminary jet experiment was performed by using air (helium substitute) inject into water (LiPb substitute) to investigate the pressure characteristics for two-pressure model of two-phase flow verification and validation. By changing the inlet air pressure (1MPa, 2MPa, 4MPa, 6MPa, 8MPa) and cover gas volume fraction (10%, 20%, and 30%), the gas injecting into water experimental results have shown that a first sharp pressure peak appeared due to “gas hammer” and the value of the first peak increased with the decreasing of cover gas volume fraction. The experimental and the calculated results showed a good agreement.
Accident analysis of helium cooled ceramic breeder test blanket system

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With China signing Test Blanket Module Arrangement (TBMA) with ITER Organization for Helium Cooled Ceramic Breeder (HCCB) Test Blanket System (TBS) in February 2014, Institute of Nuclear Energy Safety Technology (INESC), Chinese Academy of Sciences (CAS), becomes one of the leading teams undertaking its corresponding research and development, and is mainly responsible for structure material development and safety analysis. As an important part of the HCCB TBS safety assessment, accident analysis will be presented in this paper with the updated identification of reference accidents based on the approved version of preliminary safety report by IO, and more scenarios will be simulated and then analyzed using the thermal hydraulics code RELAP 5, such as the helium ingress into the HCS cooler. Uncertainty evaluation by the performance of sensitivity analyses will be used to assess the influence of variations to specific inputs/variables/models on the significant parameter results. The uncertainty analysis approach of RELAP5 is Best Estimate Plus Uncertainty (BEPU) and will be extended for HCCB TBS to provide a direct understanding of the contribution of variations to specific parameters. The primary objective of the above analysis is to evaluate the consequential radiological doses outside the ITER facility in scenarios selected to envelope all conceivable events, and thereby demonstrate compliance with the General Safety Objectives of the project. Keywords: HCCB TBS, accident analysis, RELAP
Methodology for the improvement of the AINA code wall-model applied to DEMO-WCPB blanket

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For almost ten years now, several safety studies of plasma-wall transients have been performed with AINA code for ITER, the European DEMO design (e.g. HCPB) and Japanese one (e.g. Water Cooled Pebbled Bed or WCPB) to establish an envelope for the worst effects of ex-vessel LOCA and overfuelling. For this purpose, for each blanket type a specific wall-model has been developed for different AINA code versions adapting also the plasma features if necessary. The implementation of a robust, conservative and optimized wall-model plays a key role in obtaining reliable and time achievable thermo-hydraulic results. The plasma facing components, up to the Vacuum Vessel Inner shell, are discretized in the poloidal coordinates according to the number of blanket and divertor regions. They are independent elements linked by radiation heat process. In each zone the thermal evolution (both steady state and transient) is determined and used as input to the plasma model to estimate for instance the erosion fluxes. At the same time, the materials temperatures have a not-negligible impact on the nuclear deposition (or NHD), due the density variation and the Doppler effect broadening which is being determined by means of MCNP and NJOY. The NHD has been parametrized in function of temperature, poloidal coordinates, accident type, material temperature and radial distance. The thermo-hydraulic problem is obviously iterative and very computationally demanding. For this reason, the 1D/2D models are obtained and implemented in AINA starting from a detailed 3D model. The solvers are based on multigrid algorithm over a finite volume scheme. Temperature adjusting coefficients are implemented to maintain the temperature peaks. Also, independent verifications of the wall thermal behavior have been performed using ANSYS-Fluent®. This poster aims to describe, mainly, the improvement in wall model generation process for the Japanese DEMO-WCPB options since the last AINA release.
Laser-driven accelerator of intense plasma beams for materials research
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A concept and a laboratory model of the laser-driven accelerator of plasma beams for materials research is presented. The accelerator is based on the laser-induced cavity pressure acceleration (LICPA) scheme [1] and includes four parts: (1) the laser driver, (2) the plasma cavity where high-temperature plasma is created by the laser driver and a high plasma pressure is generated, (3) the acceleration channel where the plasma beam is formed and accelerated by the plasma pressure, (4) the beam guiding channel which enables us to control the plasma beam parameters (the beam fluence, intensity and duration). It is predicted that the accelerator employing a commercial nanosecond laser of energy 2 – 3 J would be capable of producing a plasma beam of controlled composition and the beam fluence up to 200 J/cm$^2$, the beam peak intensity up to 20 GW/cm$^2$ and the beam duration within the 10 ns – 10 ms range. The accelerator has a potential to work with a repetition rate up to a few Hz (in a burst of ~ 50 - 100 shots) with the beam average intensity up to 1 kW/cm$^2$. A laboratory model of the accelerator with a 0.5J/4ns Nd:YAG laser driver was built and tested. A CH plasma beam of the fluence ~ 10 J/cm$^2$ and the peak intensity ~ 100 MW/cm$^2$ at the accelerator channel exit was produced with the laser-to-beam energy conversion efficiency approaching 15 %. A strong surface damage of various metal samples by the beam was observed. The proposed accelerator of plasma beams is a novel tool for materials research which seems to be particularly useful for testing materials proposed for future fusion reactors both the MCF and ICF ones. [1] J. Badziak et al., Phys. Plasmas 19, 053105 (2012).
Guiding of laser beam in magnetized quantum plasma

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Interaction of high power laser fields with plasma is important for many applications including laser fusion, laser wakefield acceleration and x-ray lasers. At high laser intensities, nonlinear interactions between plasma and laser becomes significant. In the last ten years, there has been a great deal of interest on plasma systems where the quantum effects are important. Consideration of quantum effects in plasma is growing exponentially because of their relevance to wide range of applications. The importance of quantum plasma in fusion has already been explored. In the present paper, we focus on the recently developed quantum hydrodynamic (QHD) model. The nonlinear paraxial wave equation having linear and nonlinear source terms, which include contributions due to ponderomotive force, quantum effects and perturbations due to the presence of uniform magnetic field alongwith relativistic nonlinearities for quantum case and an envelope equation for laser radiation has been obtained using the source dependent expansion (SDE) technique. The evolution of the spot size is derived and the effect of density perturbations on the process of self-focussing is studied.
Evolution of calorimetry methodology for beam current measurement of the LIPAc

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IFMIF (International Fusion Material Irradiation Facility) will generate 14 MeV neutron flux for qualification and characterization of suitable structural materials of plasma exposed equipment of fusion power plants. IFMIF is an indispensable facility in the fusion roadmaps since provide neutrons with the similar characteristics as those generated in the DT fusion reactions of next steps after ITER. IFMIF is presently in its EVEDA (Engineering Validation and Engineering Design Activities) phase. As part of IFMIF Validation Activities, LIPAc (Linear IFMIF Prototype Accelerator), designed and constructed mainly in European labs (CIEMAT, CEA, INFN and SCK,CEN) with participation of JAEA, is currently under installation at Rokkasho (Japan). LIPAc will accelerate a 125mA CW and 9MeV deuteron beam for a total beam average power of 1.125MW. During the beam commissioning of the currently undergoing injector characterization, the exact absolute value of the beam current is still uncertain because the interceptive beam diagnostics devices have no electron suppressor. For high intensity beam the calorimetric measurement method is employed normally, in addition to ACCT at the end of LEBT (Low Energy Beam Transport) to reduce the impact of secondary electrons on current measurements. The LIPAc has such a capability by using a number of thermocouples to measure the temperature of inlet and outlet of cooling water channels in Faraday Cup, Injector Cone and Beam Stop of LEBT. This paper describes the techniques of the calorimetric measurement method of beam current of LIPAc with the recent results and comparison to using the other measurement methods of beam current.
RF power tests of rf input coupler for the IFMIF/EVEDA RFQ prototype linac

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For the IFMIF/EVEDA accelerator prototype RFQ linac, the operation frequency of 175MHz was selected to accelerate a large current of 125mA. The driving RF power of 1.28MW by 8 RF input couplers has to be injected into the RFQ cavity for CW operation mode. For each RF input coupler, nominal RF power of 160kW and maximum transmitted RF power of 200kW are required. For this purpose, an RF input coupler based on a 6 1/8 inch co-axial waveguide was designed. The RF coupler has cooling channels into the loop antenna and inner/outer-conductors around the RF window, and the RF window module is replaceable by flanged connection. RF power tests using a high voltage standing wave on a high-Q load circuit were carried out. This circuit consists of the RF input coupler, a co-axial phase shifter and stub tuner. The RF input coupler and the stub tuner are located in both circuit edge, and these work as a short plate. Therefore, a standing wave due to incident/reflected RF power can be generated when RF power injection into this circuit. Using this standing wave, equivalent RF power of 200kW-14 sec CW operation was performed after four days of RF aging. No RF contact defects, unnecessary low-Q value and extraordinary outgassing were observed. This article describes RF power tests of RF input coupler using a high voltage standing wave for the IFMIF/EVEDA accelerator prototype RFQ linac.
6Li-D thermal-to-14 MeV neutron converter in the MARIA reactor for fusion materials related research

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Study of materials dedicated to fusion reactors is one of the most challenging tasks faced by fusion research. Unfortunately, the number of useful fast neutron sources with a proper neutron spectrum and high neutron fluence is limited. Currently, a better exploitation of the existing neutron sources, such as high flux fission research reactors or material test reactors, is necessary to develop further the fusion technology. In order to overcome some of the limitations of fission research reactors to simulate fusion conditions, a thermal-to-14 MeV neutron converter has been designed and constructed at NCBJ (Świerk, Poland). This converter can be installed in the core of the MARIA reactor. The preliminary MCNP calculations showed that the neutron energy spectrum inside the converter resembles the one expected in large fusion devices such as ITER and DEMO. This paper presents the experimental results obtained during one of the converter operating cycle. A set of activation detectors (Ti, Fe, Ni, Co, Ta, Y, Nb) and samples of real ITER materials (i.e. SS316L steels, W) have been irradiated inside the converter placed in KVIII/A channel of the MARIA reactor. The measured activities were compared against the quantities calculated using FISPACT-II inventory code. Based on the registered nuclear reactions and using SAND-II deconvolution code, it was possible to unfold the neutron energy spectrum characteristic for the selected converter location. Our previous studies showed that the higher the thermal neutron component in energy spectrum is the more difficult is to accurately determine the fast neutron spectrum. For this reason an attempt has been made to choose the most beneficial position of the converter.
Rotating tritium target for intense 14-MeV neutron source

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In order to study the neutronics of fusion reactor blankets, a program is underway at the IPR using 14-MeV neutron source. An accelerator based neutron generator is under development in which 30 mA deuterium beam will be accelerated up to 300 keV energy. It will then impinge on a rotating tritium target to producing nearly isotropic 14-MeV neutrons. The expected neutron yield is $3-5 \times 10^{12}$ n/s. The rotating target has been developed for intense neutron source. Total estimated power density on the rotating target is 11.5 kW/cm$^2$ for the diameter and power of D$^+$ beam are 10mm and 9kW (300kV,30mA). The simulation by CFD method have been carried out to investigate the heat transfer in rotating target system. In this paper, the design and analysis of the rotating tritium target system of intense neutron source is discussed and result of beam test performed using D$^+$ beam at 90keV, 20mA, 15mm beam diameter, resulting 1 kW/cm$^2$ beam power stopped at the surface of dummy copper disk 3mm in thickness and 180 mm in diameter is presented.
The LIPAc (Linear IFMIF Prototype Accelerator) is a prototype that ends in a Dump made of copper with conical shape and cooled by water moving at high speed on the outer surface. The shape of the dump is intended for a redistribution of a very high density power of the deuteron beam to be stopped (1.12 MW) leading during normal operation to reasonable temperatures and thermal stresses well below the safety margins. In case the beam reaches the dump with abnormal misalignment or offset, the local temperatures at the water-copper surface will give rise to boiling onset which can be used as a warning flag before an excessive thermal stress distribution could endanger the mechanical integrity of the Beam Dump. Previous experiments having the same goal were carried out with hydrophones to detect the onset of boiling [doi:10.1016/j.fusengdes.2015.01.011]. The current article deals with the treatment of the signal coming from an accelerometer. The article will go into detail of the advantages and drawbacks of the accelerometer versus the hydrophone as the sensor to be used. Improvements in the data treatment to refine the detection of the boiling onset will be explained as well. Real case problems like the noise coming from external or undesired sources are considered and a number of recommendations are derived for the implementation of the system in the cooling system of the LIPAc Beam Dump to be installed in Rokasho. This work has been partially funded by the MINECO Ministry under project FIS2013-40860-R and the agreement as published in the Spanish BOE (BOE n14, p. 1988).
P3.005

Thermal validation of the start-up transient scenario of the EU IFMIF target assembly concept

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The International Fusion Materials Irradiation Facility (IFMIF) is an accelerator-driven intense neutron source where fusion reactor candidate materials will be tested. The neutron flux is produced by means of a deuteron beam (250 mA, 40 MeV) that strikes a target of liquid lithium circulating in a loop. The support on which the liquid lithium flows is the most heavily exposed component to the neutron flux. The design of the target assembly (TA) system is today well advanced and a full size prototype was manufactured with the objective to carry out almost all the validation and optimization activities needed for a comprehensive final design of this component. One of the main issue to be investigated for the target system is the temperature distribution in the backplate at the start-up of the lithium loop. The flowing liquid lithium is injected in the backplate channel at 250°C and to prevent thermal shock, that may lead to rupture of the backplate itself, it has to be kept at a temperature in the proximity of the flowing lithium temperature. To address this aspect, a thermal analysis of the start-up transient scenario for the TA prototype was carried out in the past in collaboration with the University of Palermo by means of a finite element model implemented through a qualified software and an experimental test campaign has been set up for its validation at ENEA Brasimone. The optimum configuration of the heating system to allow the right temperature distribution on the backplate has been established through a continuous benchmarking between the simulation and the experimental results. Results of the experimental activities carried out together with the optimization of the configuration of the TA target heating system for the achievement of a suitable temperature distribution on the backplate are described and discussed in the paper.
Feasibility of IFMIF-DONES for other science projects

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IFMIF-DONES - a powerful neutron irradiation facility for studies and certification of materials - is planned as part of the European roadmap to fusion electricity. Its main goal will be to study properties of materials under severe irradiation in a neutron field similar to the one in a fusion reactor first wall. It is a key facility to prepare for the construction of the DEMO Power Plant envisaged to follow ITER. As part of the present deliberations about financing and siting a DONES facility, it is being considered to extend the objectives of IFMIF-DONES beyond its standard program of material studies for fusion reactors. Various scientific areas, such medical applications, nuclear physics, astrophysics, basic physics studies and industrial application of neutrons are under consideration as complementary research topics. A White Book report on “IFMIF-DONES for isotope production, nuclear physics applications, materials science and other research topics” is under preparation. The conclusions of the White Book will be presented identifying the most promising science projects that could be developed at IFMIF-DONES without compromising its main role. The possible implementation in IFMIF-DONES of these additional experiments will be also discussed. This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.
Swelling analysis and design optimization of the IFMIF target assembly with bayonet backplate

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The availability of a high flux neutron source for testing candidate materials under irradiation conditions which will be typically encountered in future fusion power reactors is a fundamental step towards the development of fusion energy. To this purpose, IFMIF (International Fusion Materials Irradiation Facility) represents the reference option to provide the fusion community with a source capable of irradiating samples at a damage rate of up to 20 dpa/fpy (in steel) in a volume of 0.5 l. This concept is based on a high-speed liquid lithium target which is stricken by a 10 MW double deuteron beam to produce 14 MeV-peaked neutrons. In the framework of the engineering design activities of IFMIF, ENEA is committed in the design of the lithium target assembly (TA) with removable (bayonet) backplate (BP) whose development has recently progressed under the IFMIF/EVEDA project up to a well advanced stage. However, an optimization of the system is still to be accomplished. In particular, the BP design needs to be revised in order to satisfy the ITER SDC-IC reference design criteria for thermally-induced stresses and fulfill the requirements on its lifetime which is limited by the neutron-induced swelling effects. In this work, a full thermomechanical analysis of the whole TA including a pseudo-transient simulation of the swelling effects in the BP over one year of full power operation was performed by the University of Palermo by means of a 3-D finite element model implemented through a qualified FE software package. A detailed neutronic analysis was also performed by ENEA using the MCNP code to obtain the prompt nuclear responses to be used as input for the thermomechanical calculations. A new BP design capable to verify the design rules criteria and ensure its required swelling lifetime is proposed and described on the basis of the results of the performed analysis.
Design, integration and manufacturing of the MEBT and DPlate support tables for IFMIF LIPAc

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The International Fusion Materials Irradiation Facility (IFMIF) aims to provide an accelerator-based, D-Li neutron source to produce high energy neutrons at sufficient intensity and irradiation volume for DEMO materials qualification. Part of the Broader Approach (BA) agreement between Japan and EURATOM, the goal of the IFMIF/EVEDA project is to work on the engineering design of IFMIF and to validate the main technological challenges which, among a wide diversity of hardware includes the LIPAC (Linear IFMIF Prototype Accelerator), a 125mA CW deuteron accelerator up to 9 MeV mainly designed and manufactured in Europe. The Medium Energy Beam Transport line (MEBT) is in charge of the beam transport at 5 MeV/125 mA and matching beam parameters between two acceleration structures, the RadioFrequency Quadrupole (RFQ) and the Superconducting RF linear accelerator (SRF Linac), while the Diagnostic Plate (DPlate) is a movable module with a set of diagnostics and instrumentations in charge of characterize the beam in the different accelerator commissioning stages (RFQ commissioning, SRF Linac commissioning) and provide accelerator operational parameters. Both beamline designs are state of art, being a real engineering (and mechanical) challenge, due to the compactness and alignment requirements from beam dynamics, and the seismic requirements from the accelerator site. An optimized design is critical in order to reduce beam losses and production of radiation at high power beam. The goal of this paper is to present the mechanical design and analysis of the MEBT and Diagnostic Plate support tables, as well as their manufacturing solutions and mechanical integration with the components installed both in the MEBT and Diagnostic Plate. The mechanical design and integration will show the engineering development, adopted to fulfill the strict structural, seismic and alignment requirements. This work has been supported by Spanish government (MINECO) in the frame of the BA Agreement Activities.
P3.009

The Related Experimental Issues of PbLi Coolant by the Effect of Magnetic Field

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Because of the depletion and limitation of natural energy sources, fusion energy is the promising and irreplaceable way for energy development in the future. As the only energy conversion unit in the fusion reactor, PbLi blanket is considered as one of the important blankets for DEMO and fusion reactors. Lead Lithium (PbLi) is designed as tritium breeder, neutron multiplier and coolant. Before the engineering application of fusion energy, series of issues need to be validated completely out of pile, such as corrosion behavior of blanket structural materials, magnetohydrodynamic (MHD) effect for PbLi fluid, safety issues of coolant when heat exchanger breaks, and purification technology of PbLi alloy, etc. Dual Functional Lead Lithium (DFLL) blanket is designed as one of the DEMO blanket in China, and a multi-functional PbLi loop DRAGON-IV was built to study the R&D issues of DFLL blanket technologies. In order to study the MHD effect, the pressure drop was tested by pressure differential meters in the test section under 2T magnetic field, which was validated by MTC code, one validation MHD simulation software developed by FDS team. The corrosion experiment of CLAM steel, the structural material of China DEMO blanket, was carried out in the flowing PbLi under 1-2T magnetic field for a thousand hours. The confirmed experiment of impurities type and quantity in PbLi alloy was implemented preliminarily, which was very important for the purification system design and application in future. The above experimental results were achieved to support the development of the blanket system, too. In this manuscript, the details of DRAGON-IV was introduced, the related experimental issues of PbLi coolant were elaborated, and the experimental results were discussed for further application, which support strongly the design and development of engineering technologies for China PbLi blanket. Keywords: DFLL blanket; DRAGON-IV; MHD effect; Corrosion; Impurity type
P3.011

Water Evaporation - Condensation Cooling System Design for Pb-Li17 Cold Trap

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The liquid metal eutectic Pb-Li17 is considered as one of the possible coolants for the blanket of the fusion reactor DEMO. The main reason for usage of the eutectic Pb-Li17 is the Tritium breeding. The eutectic flow separates alloys of the structural steels and thus be the cause of them corrosion. The cold trap is a device for corrosion products removing from liquid metal. The cold trap was developed by the Research Centre Rez within the Centre for Advanced Nuclear Technologies and consists of three loops. First one is for eutectic flow. The secondary loop contains water-steam mixture which provides cooling of the primary loop by water evaporation. The accumulated thermal energy is subsequently transferred by condensation to the tertiary loop - water cooler (condenser) which performs final heat removal from the trap body. Separation of corrosion products is carried out by gravity when the flow of eutectic with impurities is distributed by technological membranes. The process of separation occurs at lower temperatures than the operating conditions of eutectic in the fusion reactor blankets. For this reason, an efficient cooling of the liquid metal in the cold trap device is necessary. As one of the acceptable variants is the contemplated cooling by water evaporation in the secondary loop and subsequent condensation of the created water–steam mixture at the tertiary loop (condenser). Adding inert gas (Argon) into the mixture is necessary to achieve and control required higher saturation conditions. The presence of this gas in the mixture significantly influences the behaviour of heat and mass transfer. It brings an effect to heat removal from cold trap body. For solving these processes 1D model is developed and applied. Achieved results are analysed and discussed.
The Demonstration Fusion Power Reactor (DEMO) is supposed to be the step in between ITER and the first commercial fusion power plant. In the framework of one mission of the “Work plan for the roadmap to fusion energy 2014-2018” a work package Tritium, Fuelling and Vacuum (TFV) was launched. As part of this project, the examination of requirements for the matter injection system is ongoing covering all aspects of plasma operation like pre-fill, ramp-up and steady state including plasma enhancement gases. In a first step, requirements for keeping steady state burning plasmas were elaborated. Related modelling activities indicated that only sufficiently deep fuel deposition can achieve target operational parameters. Hence, suitable techniques had to be identified and evaluated with respect to their availability and capability. Finally, cryogenic pellet injection was chosen as the most realistic option for core fuelling of the plasma. From modelling activities, assuming for the pellet mass the ITER reference value, required launching speeds were derived, with respect to different injection geometries. Several techniques for pellet injection have been benchmarked in view of the defined requirements. Gas puffing and the respective technical system are necessary for pre-fill, ramp-up and plasma confinement enhancement. The ITER GIS is assessed in view of suitability for DEMO. The tubing system and the manifold concept can be adopted. The Gas Valve Box (GVB) is considered not to be an optimum solution for DEMO. Instead of this GVB, a pressure based RUN/VENT flow regulation and injection system is proposed in order to meet DEMO requirements. The principle of this system is described as well as some considerations about injection locations; further orienting gas flux numbers are provided. The next step in the TFV sub project “matter injection” will be to work out a conceptual design for the pellet injection system.
Overview and status of construction of ST40

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Recent advances in the development of high temperature superconductors (HTS) [1], and encouraging results on a strong favourable dependence of electron transport on higher toroidal field (TF) in Spherical Tokamaks (ST) [2], open new prospects for a high field ST as a compact fusion reactor or a powerful neutron source [3]. The combination of the high beta (ratio of the plasma pressure to magnetic pressure), which has been achieved in STs, and the high TF that can be produced by HTS TF magnets opens a path to lower-volume fusion devices, in accordance with the fusion power scaling proportional to $\beta^2 2B_t^4 Vol$. The new generation high field spherical tokamak ST40 ($R_0=0.5-0.7m$, $A=1.7-2.0$, $I_{pl}=2MA$, $B_t=3T$, $k=2.5$) is under construction by Tokamak Energy Ltd, UK, with the first plasma expected in 2016. Overview of the ST40 project will be presented. Main objectives of the project, parameters of the tokamak, physics programme issues will be described and physics and engineering challenges (mainly connected with the high toroidal field and high wall and divertor power loads) of this device will be discussed. The device is aimed to demonstrate burning plasma parameters (nTtauE) with a possibility of DT operations. The present status of the construction will be reported. The demonstrate on of reliable operations of a compact high field ST, with the toroidal field up to 5 times higher than in presently operating STs, will significantly advance ST research. ST40 will be a step in the commercial exploitation of Fusion for the development of the Fusion Energy. [1] M GRYAZNEVICH et al., Fusion Eng. and Design, 88 (2013) 1593. [2] M. VA LoviC et al., Nuclear Fusion, 49 (2009) 075016. [3] M GRYAZNEVICH et al., Fusion Science and Technology 61 (2012) 89
P3.014

Manufacturing of a small scale Stellarator in Costa Rica

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The manufacturing methods and issues found during the construction of the Stellarator of Costa Rica 1 (SCR-1) will be discussed. The SCR-1 is a small modular stellarator developed by the Instituto Tecnológico de Costa Rica (ITCR). Currently, it’s being tested for the first plasma discharge. SCR-1 is a 2-field period small modular stellarator (Ro=0.238 m, =0.054 m, Ro/a>4.4, plasma volume =0.01 m³). Constructed from a 6061-T6 aluminum torus-shaped vacuum vessel with 10 mm thickness and a volume of 0.0418 m³ [1]. Plasma will be confined by magnetic field with a strength of 43.8 mT on axis. This will be generated by 12 copper modular coils with 4.6 kA-turn each. The SCR-1 plasmas will be heated by ECH 2nd harmonic at 2.45 GHz with a plasma density cut-off value of 7.45 × 10¹⁶ m⁻³. Two magnetrons with a maximum output power of 2 kW and 3 kW will be used during plasma shot. For its construction, a combination of classic and modern manufacturing techniques was used. Its main body was CNC machined from two prismatic aluminum blocks. The vacuum vessel has 24 ports: 22 circular CF ports, and two rectangular ports which were mechanized according to in house design dimensions. The coil supports were obtained from a 3D printed and casted mold. This was preferred given their complex geometry and the modularity needed for their assembly. MIG welding was used to join all the previous elements. Verification testing on mark positioning were performed where coil support were placed. Once the coil supports were positioned, the copper wire was allocated between the coil supports by an experienced and certified technician. Advantages and disadvantages of manufacturing methods of vacuum vessel, CF ports and modular coils will be presented.
SST-1 up-gradation update & recent experiments in SST-1


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Steady State Superconducting Tokamak (SST-1) at Institute for Plasma Research is a ‘working’ experimental superconducting device since late 2013. SST-1 has been upgraded with Plasma Facing Components and is getting prepared towards long pulse operations in both circular and elongated configurations. Initial experiments have begun in SST-1 with circular plasma configurations. SST-1 offers a unique possibility of investigating long pulse discharges with large aspect ratio (> 5.5) compared to contemporary devices. Presently, SST-1 standard ohmic discharges are in excess of 100 KA with typical core density $\sim 2 \times 10^{19} \text{ m}^{-3}$ and core electron temperatures $\sim 500 \text{ eV}$ having duration in excess of 300 ms. A 42 GHz ECR pre-ionization source at $\sim 150 \text{ KW}$ in 1.5 T central field breaks down the gas, the current starts up at $\sim 1.3 \text{ MA/s}$ in 60-80 ms in an induced field of $\sim 0.3 \text{ V/m}$. These standard discharges demonstrate copious saw teething and MHD activities as the pulse progresses including NTM, mode locking and MHD characteristics. PFC equipped SST-1 has completed these basic experimental studies confirmed with simulations. These includes eddy currents influencing the NULL dynamics, field errors, equilibrium index evolutions, wall influencing plasma characteristics, plasma positions, plasma rotational and Tearing Mode characteristics including the island width and island growths etc. Presently, SST-1 is attempting at multi second long high aspect ratio plasma discharges by coupling the Lower Hybrid with the Ohmic plasma as well as with robust real time density and pressure controls. The up-gradation details, salient early plasma characteristics in large aspect ratio PFC equipped SST-1 plasma and future experimental plans towards long pulse operations in SST-1 will be elaborated in this paper.
This paper presents the results of a study that was performed on conceptual solutions for assembly and handling of EC components inside the EC upper and equatorial port cells. Particular topics that are discussed include the access to the waveguides and auxiliary feedthroughs of the launchers at the port plug closure plate, (dis-)assembly & alignment of the ex-vessel waveguide in the port interspace via a ceiling support structure in the port cell and procedures for handling and alignment of the individual segments of the first confinement ex-vessel transmission lines. The ITER Tokamak will become activated over time, leading to non-negligible shutdown dose rates, mainly inside the port cell interspaces. To minimize dose uptake for maintenance personnel, ITER requires that the design of ex-vessel equipment is optimized to keep maintenance time short and exposure levels ALARA (as low as reasonably achievable). In order to achieve this, some of the design principles and guidelines for Remote Handling (RH) compatibility are applied to ex-vessel EC components. This helps to ensure that all dismountable interfaces have good accessibility both by man and machine and that all maintenance tasks are well documented, rehearsed and can be demonstrated either through Virtual Reality simulation or through hardware mock-up.
Mechanical and seismic analyses of the ITER Electron Cyclotron Upper Launcher First Confinement System

MAS SANCHEZ, Avelino; AIELLO, Gaetano; CHAVAN, Rene; GAGLIARDI, Mario; GOODMAN, Timothy; HENDERSON, Mark; LANDIS, Jean-Daniel; SAIBENE, Gabriella; SANTOS SILVA, Phillip; SUDKI, Bassem; VACCARO, Alessandro

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The Electron Cyclotron Upper Launcher (ECUL) is an eight beamline ITER antenna aimed to drive current locally inside the islands that may form on the q= 3/2 or 2 rational magnetic flux surfaces in order to stabilize neoclassical tearing modes (NTMs). The primary vacuum boundary at the port plug extends into the port cell region through the ex-vessel mm-wave waveguide components, defining the so-called First Confinement System (FCS). Each transmission line considered here, designed for the transmission of 1.5 MW of mm-wave power at 170 GHz, is delimited by the closure plate at the port plug back end and by a diamond window in the port cell. The FCS essentially consists of a Z-shaped set of straight corrugated waveguides connected by miter bends with a nominal inner diameter of 50 mm. Thermal expansion, seismic events and plasma loads result in displacements of the vacuum vessel, relative to the tokamak building, that are transferred to the FCS at its interfaces with the port plug. The thermal expansion arising from ohmic losses in the transmission line, water cooling and inertial loads contribute additional displacements within the FCS. In absence of suitable inline waveguide bellows, the adaptation to such imposed displacements is provided by bending compliance of the straight waveguide sections. This paper describes work related to the selection of the applicable load combinations for the FCS, as well as the mechanical and seismic analyses carried out to assess the performance of the system against these load combinations. This global analysis provides load/displacements inputs for component 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.
P3.020

Thermal mechanical analyses of mm-waveguide cooling concepts for the ITER ECHUL first confinement system

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The ITER Electron Cyclotron Heating Upper Launcher (ECHUL) will be used to drive current locally inside magnetic islands located at the q=2 (or smaller) rational surfaces in order to stabilize neoclassical tearing modes (NTMs). Each antenna consists of eight beam lines that are designed for the transmission of up to 1.5 MW of mm-wave power at 170 GHz. The First Confinement System (FCS) is formed by the ex-vessel mm-wave waveguide components, for which SIC-1 classification requirements apply. Each transmission line consists in a Z shaped set of straight corrugated aluminum alloy (EN AW-6061) waveguides connected by miter bends with a nominal inner diameter of 50 mm. In addition to the ohmic losses related to the mm-wave transmission, the waveguides of the FCS shall be capable of resisting the applied external loads and displacements, and also operate under thermal cyclic loading during ITER operation. The FCS waveguide nominally transmits up to 1.5 MW of CW170 GHz mm-wave power, with at least 90 % of the power in the main HE11 mode. While actual losses will have to be determined experimentally, estimated losses are considered additive and mm-wave power is assumed to be converted into heat by ohmic dissipation in the waveguide, with intensity peaks reaching up to 9000 W/m². For continuous working operation at nominal transmitted power, temperature control of the waveguide is required via an active cooling system. Available commercial solutions for the waveguide are incompatible with the FCS, as they will be subject to higher heat fluxes and shall comply with ITER SIC-1 requirements. Therefore a dedicated cooling system must be designed. This study presents the results of the thermal mechanical analyses of three different cooling concepts, and concludes which is the most suitable for the final FCS system design.
Characterization of metallic seals used in the waveguide flange coupling of the ITER ECHUL

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The ITER Electron Cyclotron Heating Upper Launcher (ECHUL) will be used to drive current locally inside magnetic islands located at the q=2 (or smaller) rational surfaces in order to stabilize neoclassical tearing modes (NTMs). Each antenna consists of eight beam lines that are designed for the transmission of up to 1.5 MW of mm-wave power at 170 GHz. The First Confinement System (FCS) is formed by the ex-vessel mm-wave waveguide components, for which SIC-1 classification requirements apply. Each transmission line consists in a Z shaped set of straight corrugated aluminum alloy waveguides connected by miter bends with a nominal inner diameter of 50 mm. The FCS system also encompasses intrinsic components such as: closure plate, mm-waveguide taper, isolation valve, diamond window and an EU-US interface waveguide. All FCS components are connected via a dedicated flange coupling, with a bolted connection, comprised of two concentric metallic seals in order to satisfy the Safety and Vacuum quality requirements. The coupling shall be capable of resisting the applied external loads and displacements, including thermal cycles due to ohmic losses in the transmission line, and also adhere to the mating of different materials (EN AW-6061, CuCrZr, SS 316LN) while maintaining ultra-high vacuum tightness. The present study uses a dedicated experimental apparatus for the characterization of the mechanical and vacuum properties of a single metallic seal under compression, with applied heat flux. In order to ensure reproducibility of the components mechanical characteristics, randomly chosen metallic seals of each diameter will be tested from two independent manufactures' (A and B). This work was supported in part by the Swiss National Science Foundation. This work was carried out within the framework of the ECHUL consortium. The experimental apparatus and metallic seals were financed via the F4E-OPE-528 contract and graciously made available to be used in this study.
P3.022

Development of a direct mirror angle detector for ITER EC launchers

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The new mirror angle detector for ITER EC launchers, applying a rotary capacitor, a RF feeder, RF circuits and several hundreds MHz RF has been developed. The rotary electrode is attached to the rotation axis of the mirror and the stationary electrode is connected to a RF feeder. The reflected RF wave at the rotary capacitor comes back to the feeder and phase of the reflected RF wave changes depending up on capacitance due to the overlap condition between the two electrodes. The phase change tells us the rotation angle of the mirror. It has been difficult to cool the stationary electrode that must be electrically isolated. Then, application of a co-axial choke stub to the support of the stationary electrode has been utilized since it acts as the electrical isolator. As a result, cooling water can be supplied to the stationary electrode through the center conductor of the choke stub. To solve this issue, a co-axial choke stub tuned at designated RF frequency is applied for for the support of the stationary electrode since it acts as a notch filter to electrically isolate it. The angle monitor was designed as the radius of the rotary and stationary electrodes, the electrodes’ gap, the stub length and RF frequency were 50mm, 42mm, 2m, 78mm and 900MHz, respectively to verify the applicability to the EC launchers. Then, the mock-up were fabricated and tested. It was obtained that phase difference of the reflected RF wave at the rotation angle change of 14° was 26°, which was well agreed with the design value of 25.7°. It is concluded that the new mirror angle detector is applicable to the rotatable mirror of the EC launchers.
Demonstration of synchronous control of EC TL switch and gyrotron for ITER EC system

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The Electron Cyclotron Heating and Current Drive system developed for ITER is made of 12 sets of High Voltage Power Supplies, 24 Gyrotrons, 24 Transmission Lines and 5 Launchers, 4 UL located in upper ports and 1 EL at the equatorial level. The ITER operation requires to switch operating launcher during the plasma operation with short interval, namely mid-pulse switch operation. To change the waveguide switch which directs RF power either to upper port or equatorial port launchers, the gyrotron has to stop RF power during the switch operation since the direction is changed by mechanical movement of mirror position which takes a few seconds. Since ITER EC system is based on multi-subsystem concept, each subsystem has its own subsystem control unit (SCU) and EC main controller supervises all subsystem controllers. Hence cooperative operation requires to share the information of both RF power status and switch status between gyrotron SCU and TL SCU via main controller. The design of inter-subsystem control scheme is a key issue for ITER EC system control and its evaluation is required. In JAEA, gyrotron and ITER relevant TL test stand were utilized for demonstration of mid-pulse switch operation. For this purpose, SCUs for each subsystem and main controller were developed using ITER relevant control system. The operation of mechanical switch during gyrotron pulse was demonstrated. During the 150 s operation of high power gyrotron at 400 kW level, waveguide switch in TL was operated to change the direction of RF power. The time duration for switch operation with inter subsystem control scheme took 1.5 s in total. The synchronizing of RF power suspend and resume with switch motion was succeeded and RF power direction control during the gyrotron operation was successfully demonstrated.
Overview of the EU HV Power Supply System for the ITER EC System

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The power supply for the EC Heating system (ECPS) of ITER provides the electrical power to the 170GHz/1MW Gyrotrons. The required electrical power for the gyrotrons is not only very high but has to comply also with highest quality requirements. This paper gives an overview of the Ampegon ECPS system procured by F4E. It describes the technical requirements of the EC Power Supply system ECPS and explains how this challenging requirements can be met. The use of the Ampegon PSM and EPSM technology is justified and the expected performance of the system is shown. The ECPS of the European contribution is currently under development; the project has started by the end of 2013. An outline of the project schedule and the actual status is shown.
Experimental verification of the European 1 MW, 170 GHz industrial CW prototype gyrotron for ITER

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The EU 1 MW, 170 GHz gyrotron with hollow cylindrical cavity has been designed within EGYC (European GYrotron Consortium) in collaboration with the industrial partner Thales Electron Devices (TED) and under the coordination of Fusion for Energy (F4E). In the frame of the EU program the short-pulse (SP) version of this tube has been designed and manufactured by KIT in collaboration with TED. The experimental verification of the SP gyrotron prototype has been successfully completed in 2015. The achieved experimental results show a very stable gyrotron operation with RF output power above 1 MW at reasonable interaction efficiency around 35% (without depressed collector). The gyrotron was operated up to 10 ms pulse length; the nominal cavity mode TE32,9 has been excited at the frequency 170.1 GHz being in agreement with the ITER specification. The Gaussian mode content of the output RF beam was about 98% and the total level of internal stray radiation in the range of 2-3%. The manufacturing of a first industrial continuous-wave (CW) prototype gyrotron, based on the design of the SP gyrotron, has been completed in November 2015. The tube has been delivered to KIT and recently installed into the superconducting gyrotron magnet. The start of the tests is scheduled for February 2016. At first, in order to optimize the gyrotron operating point in terms of stability and efficiency of the RF power generation, the gyrotron will be operated in the short-pulse regime. Furthermore, detailed investigations of the mode converter efficiency, by measuring of the fundamental Gaussian mode content and stray radiation level, will be performed as well. The next phase will consist in pulse length extension and optimisation of the associated operating point. The experimental results will be presented and discussed in this paper.
Update of the cooling design if the ITER EC upper launcher

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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. The structural system of these launcher antennas will be installed into four upper ports of the ITER vacuum vessel. During operation the port plug structure will be heated by nuclear heating from neutrons and photons and thermal radiation from the plasma. Also stray radiation and power losses from the MW-system can create local heating of the port plug structure. This is why the port plugs must be equipped with a powerful heat removal system based on cooling water circuits. Beside reliable dissipation of up to 600 kW total heat applied to the port plug also good flow characteristics, adequate distribution of coolant in parallel branches and steady temperature gradients between interacting cooling water channels must be ensured by proper layout of the cooling system. This paper outlines an update of the cooling system according to design changes of the EC upper port plugs induced by the latest blanket geometry specifications. All relevant results of thermo-hydraulic analyses for different operation scenarios and fault conditions are presented as well as the thermo-mechanical behavior and manufacturing aspects. Acknowledgement: This work was supported by Fusion for Energy under the grant contract No. F4E-2010-GRT-161. The views and opinions expressed herein reflect only the author’s views. Fusion for Energy and ITER are not liable for any use that may be made of the information contained therein.
The Tokamak à Configuration Variable (TCV) has been recently equipped with a 1 MW neutral beam heating (NBH) injector\textsuperscript{1}. Two new stainless steel ports with rectangular aperture of 170x220mm have been manufactured and installed for this purpose. The NBH injector is connected to one of them via a stainless steel port extension. The port and its extension together form the beam duct between the vacuum vessel (VV) and the injector. A preliminary thermal analysis of the beam duct showed no expectation of thermal events such as overheating. Indeed, although the beam power flux near the internal faces of the beam duct reaches a maximum of 2 kW/cm\textsuperscript{2}, the very grazing incidence angle was expected to reduce the effective wall flux by an order of magnitude. As a result, the design and manufacturing of the beam duct did not include any provision for cooling. However, early in 2016 the commissioning of the NBH injector showed high overheating of the port extension, resulting in local melting and ultimately loss of vacuum insulation. This paper describes the thermal measurements and improvement of the installed uncooled beam duct and the design, analysis, manufacturing and installation of a new beam duct with an integrated cooling system. This work was supported in part by the Swiss National Science Foundation. \cite{1} A. N. Karpushov et al., Neutral beam heating on the TCV tokamak. This conference.
Commissioning of the heating neutral beam injector on the TCV tokamak

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The TCV tokamak infrastructure has been recently adapted to leave access for a neutral beam (NB) injector capable of 1MW of neutral power during 2sec into the TCV plasma. BINP has been in charge to design and to procure this equipment, taking care of the experimental constraints imposed both by the future physics objectives of TCV, as by the mechanical requirements complying with the tight space available for installing the material inside the tokamak building. The development and design phase, as the preparation work have been described in [1]\textsuperscript{1} & [2]\textsuperscript{2}. This paper will focus on the commissioning description of the NBH system at the SPC site and on the presentation of the main results obtained during the first operation phase with TCV. The steps followed from the installation of the main auxiliaries (HVPS, RF PS, cooling circuits, cryogenics equipment etc.) up to the final acceptance tests on the plasma targets will be detailed. The measurements and the associated protection interlocks acting on the TCV control system will be described, including also the human safety rules imposed to comply with a safe operation of the injector. To conclude, the further stage to finalize the integration of the NBH on TCV will be briefly developed. This work was supported in part by the Swiss National Science Foundation.

EMC improvement of an ECH power supplies system at TCV

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Three RHVPSs (Regulated High Voltage Power Supplies, 84kV/80A/2s) are installed and operated at the Swiss Plasma Center for almost twenty years. Each RHVPS supplies a cluster of three gyrotrons. Two clusters are composed of diode type gyrotrons operating at the second harmonic of the TCV electron-cyclotron frequency (X2, 84GHz), whereas the third is a cluster of triode type gyrotrons operating at the third harmonic (X3, 118GHz) [1]. During the ECH HVPSs design and installation period, grounding was mainly considered from the safety point of view. EMC (Electro Magnetic Compatibility) was not a major concern; so, the ECH plant operation was producing important electromagnetic interferences. In previous work, the MPS (Modulator Power Supply), which controls the anode to cathode voltage of X3 gyrotrons, was upgraded to improve its output voltage quality [2]. A new control system was also implemented [3]. The upgrade aimed to allow operating the X3 gyrotrons in a more reliable way and to achieve real-time X3 RF power tuning. Subsequently, it has been necessary to understand why operation of the X3 cluster was not successful (occurrences of mode jumps or arcs in the gyrotrons) when its RHVPS was driven by an external voltage reference, whereas the X2 clusters have always been able to operate in this way. This paper focuses on the EMC problems inherent to power converters. The mechanism by which these converters influence their environment is explained. Solutions to minimize the emission of interferences are given. Finally, first results of real-time X3 RF power tuning during TCV shots are shown. These improvements make possible extended TCV heating scenarios. [1] D. Fasel et al., Proc. 19th SOFT, September 1996, p. 569-572 [2] U. Siravo et al., Fusion Engineering And Design, vol. 96-97, October 2015, p. 597-601 [3] J. Dubray et al., this conference
Neutral beam heating on the TCV tokamak

KARPUSHOV, Alexander N.; CHAVAN, Rene; CODA, Stefano; DAVYDENKO, Vladimir I.; DOLIZY, Frederic; DRANITCHNIKOV, Aleksandr N.; DUVAL, Basil P.; IVANOV, Alexander A.; FASEL, Damien; FASOLI, Ambrogio; KOLMOGOROV, Vyacheslav V.; LAVANCHY, Pierre; LLOBET, Xavier; MARLETAZ, Blaise; MARMILLOD, Philippe; MARTIN, Yves; MERLE, Antoine; PEREZ, Albert; SAUTER, Olivier; SIRAVO, Ugo; SOROKIN, Aleksey V.; TOUSSAINT, Matthieu

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The TCV tokamak contributes to physics understanding in fusion reactor research based on a wide experimental tool set: flexible shaping and high power electron cyclotron heating. Plasma regimes with high plasma pressure, a wide range of temperature ratios and significant populations of fast ions are now attainable by a TCV heating system upgrade. In the first stage of the TCV upgrade program, a 1 MW neutral beam of Deuterium or Hydrogen was installed (final acceptance, early 2016), and is reported in this paper. Recently, during commissioning of the first injector (NBI) [3], 1 MW of power was delivered into plasma, at energies ranging from 15-26 keV. Record ion temperatures of 2 keV and toroidal rotation velocities up to 160 km/s were promptly observed within a few discharges. A plasma box with up to 40 kW of RF power at ~4 MHz forms the NBI plasma source. A slit-geometry in the elementary cell of injector ion optics was adopted; high aspect ratio slits, orientated in the horizontal plane, distributed inside a 250 mm radius circle. To focus the beam to the design location (~3.6 m from the ion source), the grids consist of spherical segments. An angular divergence of 20 mrad across and 12 mrad along the slits was achieved. Finally, the NBI control and power supply systems are described that allow variation and fast modulation of the NBI power. This work was supported in part by the Swiss National Science Foundation. [1] A. Fasoli for the TCV Team, Nucl. Fusion 55 (2015) 043006 [2] A N. Karpushov et al., Fusion Engineering and Design 86 (2011) 868
P3.031

Fault Detection System for ICRF Transmission Line in LHD

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The transmission line is one of the most important parts among the ion cyclotron range of frequencies (ICRF) heating devices. In the case of unwanted troubles on the line, immediate power-off is necessary for the protection of the line and for safety. In the Large Helical Device (LHD), though the causes were unclear, several troubles such as melting sometimes occurred on the line between the Final Power Amplifier (FPA) and the impedance matching device. The impedance matching device is located in the LHD hall because the shorter distance between the ICRF antenna and the matching device is better for reducing the power loss. As a result, a long distance of more than 100 m between the FPA and the impedance matching device is necessary. Therefore, it will be difficult to check the whole line temperature with IR cameras. Moreover, it is difficult to recognize the fault by the reflection power since the matching device reduces the reflection. For these reasons, we developed the fault detection system for the ICRF transmission line in LHD by applying the Scattering Matrix Arc Detection System (SMAD) technique in the JET ITER-like ICRF antenna. Three signals are combined with power combiners so that the combined signal is zero. Balancing must be maintained with arbitral output impedance. Adjustment of phase shifters and attenuators is done by changing the output impedance with the matching device. If the three signals are not balanced, the combined signal is not zero. In this case, there is a fault somewhere in the transmission line since the S-matrix has been changed, and the ICRF power must be turned off immediately. The fault detection system will be operated in the next ICRF heating experiments.
Hybrid-order nonlinear resonant interaction between an Alfvén wave and ions

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The heating of ions by an obliquely propagating shear Alfvén wave at frequencies a fraction of the particle cyclotron frequency is demonstrated analytically. Under consideration of the small wave amplitude, the resonance conditions in the laboratory frame are systematically derived by multi-scale expansion method. It is found that 1) the cyclotron resonance condition may occur at any wave frequencies. 2) the high-order cyclotron resonance conditions are satisfied, which yield higher order cyclotron resonance. In other words, the low order cyclotron resonance heating efficiency is greater than the high. 3) a navel generation mechanism on stochastic motions of ions, put forward, is that hybrid-order ion cyclotron resonance caused by a single Alfvén wave can randomize the ion motion. This phenomenon may have relevance for the heating of ions in some toroidal confinement fusion devices as well as for ion heating in the solar corona.
The efficiency of heating and current drive systems is the key for a successful operation of fusion demonstration power plants like DEMO. In an earlier review article, overall efficiencies of H & CD systems were estimated at 20 – 30% [1]. In this paper we present a breakdown of the overall efficiency for ICRF (ion cyclotron range of frequencies): 1) the technical efficiencies; 2) the interface efficiency (hardware/plasma), and 3) the efficiency power absorption in the central plasma named “heating efficiency”. The technical efficiency of the generators includes all the subsystems like power supplies, air supply, CODAC, vacuum system, etc. and is today around 60%. The pros and cons of solid-state amplifiers and the possible efficiency gains are reviewed. Losses for the matched and unmatched sections of the transmission lines are quantified. Losses in the antenna as a function of coupling impedance and antenna characteristics are discussed. The interface efficiency is the ratio of the power to the plasma inside of the separatrix to the total power leaving the antenna. Power to the edge plasma is thus counted as loss. Most heating and current drive scenarios aim at strong absorption, coupling the power to the plasma core. In the high density, high temperature plasmas of large machines such as DEMO almost all wave power is absorbed in the plasma. Whether the power centrally absorbed ends up in bulk plasma heating (“heating efficiency”) or driven current depends on the discharge parameters and the heating/current drive scenarios. Experimental evidence in present machines shows that in heating scenarios the efficiency is 70-90%. This leads to an overall efficiency for heating in the range 40% to 55%. We finally address the issue of current drive efficiency. [1] Pamela et al. Fusion Engineering and Design, 84 (2009) 194-204
Design of an ICRF system for plasma-wall interactions and plasma production studies on TOMAS

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Ion cyclotron wall conditioning (ICWC) is being developed for ITER as a baseline conditioning technique in which the ion cyclotron heating and current drive system will be employed to produce and sustain the current-less conditioning plasma. The TOMAS project (TORoidal MAgnetized System, operated at the FZ-Juelich, Germany) proposes to explore several key aspects of ICWC. This project stands on two pillars featuring plasma and material studies: (a) plasma-induced material modification and optimization of the wall conditioning efficiency via exposure of probes made of real PFC and the use of tracers; (b) detailed research on ICWC plasma production and optimisation to benchmark codes. The ICRF system requirements to fulfill the above aims are: (a) ability to couple up to 6 kW of RF power to low density and low temperature plasma ($10^{11}$ cm$^{-3}$, 3-10 eV) (b) ability to initiate plasma in broad frequency range (15 to 45MHz) for plasma production studies. For this purpose we have designed an ICRF system made of a single strap antenna within a metallic box, connected to a feeding port and a pre-matching system. We discuss the design work of the antenna system with the help of the commercial electromagnetic software CST Microwave Studio. The simulation results for a given geometry provide input impedance matrices for the two-port system. These matrices are afterwards inserted into various circuit models to assess the accessibility of the required frequency range. The sensitivity of the matching system to uncertainties on plasma loading and capacitance values is notably addressed. With a choice of three variable capacitors we show that the system becomes resilient to such uncertainties. We also demonstrate that the system can cope as well with the high reflected power levels during the short breakdown phase of the RF discharge.
Microwave studio simulation results for two NSTX HHFW antennas in a test stand

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Experimental results have shown that twelve-strap HHFW operating at 30 MHz can provide significant plasma heating for NSTX. In this case, it is important to understand the interactions between return currents on the antenna enclosure sidewalls/septa and the launched $k_{\parallel}$ spectra. CST Microwave Studio is applied to this problem with the view toward optimizing the antenna coupling to the desired spectra. Two original NSTX antenna structures, each of which includes a single feedthrough strap and its enclosure with Faraday shield, were simulated in a test stand environment. The test stand itself is a small stainless steel vacuum chamber that accommodates two antenna structures. When they were driven out-of-phase with a 0.8-sec 60-kW pulse, the infrared camera images, which imply the return currents via ohmic heating, indicate very good agreement with the simulation results for current distributions on Faraday shields. With this validation for the model, new simulations have been designed to understand the current distribution on the plasma surface by placing a copper or a stainless steel plate 5 cm away in front of the straps to mimic the plasma. Then these two antenna structures were driven out-of-phase and in-phase to show the induced current patterns on the plate. Two Faraday shield designs were used in the simulations. One has side walls that extend out to near the radius of the current straps and another has side walls that are slotted well back from the current strap radius. The simulation results clearly indicate that the interactive areas on the plate simulating the plasma surface are reduced when there are shallow slots on the enclosure side walls. This suggests that the excitation spectrum is strongly affected in the shallow slot case and hence there is likely a reduction in coupling to the desired waves in plasma as well.
Establishing the Wendelstein 7-X steady state control and data acquisition system

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The W7-X steady state control and data acquisition system has been successfully commissioned and well established to investigate plasma break down and run the first more complex physics programs during the initial operation phase of W7-X. Already in the first weeks of plasma operation, experiment programs with up to 10 minutes containing a series of up to 20 plasma discharges have been run routinely. The main components of W7-X and a list of operational diagnostics have been integrated into the CoDaC system. The control system allows setting up experiment programs using physics oriented parameters which will be transformed to their corresponding technical values. While editing, programs are checked for constraints known in advance. Before and during execution all components report their ability to run the announced program parts as basis for the program’s online feasibility check. Program progress, execution information and components’ states are monitored. Program parameters and execution status are stored with a dedicated program label in the W7-X archive. The data acquisition system continuously acquires both operational machine data and diagnostics data. Trending data can be monitored online. All data is stored with experiment-wide unique timestamps in the W7-X archive, accompanied by parameters and signal descriptions to ensure their traceability. Archived data can be browsed by time interval and signal address. For data analysis routines, all data can be accessed using the dedicated signal access programming interface or a convenient web service. Analysis results as well as measured data from diagnostics not yet integrated into the CoDaC system can be uploaded into the W7-X archive and accessed by the same interfaces. All parts of the CoDaC system have been already in operation during the W7-X commissioning phase assisting commissioning and test of components and diagnostics. The data acquisition system has been running reliably 24/7 for almost two years.
Experiences with the Segment Control system at Wendelstein 7-X operation

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Wendelstein 7-X (W7-X) is a superconducting stellarator undergoing the first experimental campaign after its commissioning. It’s characteristic feature is the steady state operation of the magnetic field. After an upgrade to cope with permanent heat loads of several Megawatts, W7-X will be able to run steady state discharges, too. This requires a control system that differs from the commonly used shot based systems. The W7-X control system supports all types of experiments compatible with steady state magnetic field operation, i.e. short plasma discharges and sequences of discharges with arbitrary time intervals. Moreover, it can handle plasma discharges of unlimited duration. This is achieved by dividing the experiment duration into time slices which are called “segments”. The use of segmented experiment programs helped cleaning the device by repetitive Electron Cyclotron Resonance power pulses with several discharges in one experiment. Using these features more than 1300 discharges have already been carried out during the first month of operation. The hierarchical layout of the control system reflects the structure of the experimental device. Each technical component and each diagnostic system including its data acquisition has its own segment control system permitting autonomous as well as co-ordinated operation. The activity of these devices is co-ordinated by a master controller during the experimental sessions. Component activities can be edited and tested by the component experts in parallel and later inserted into the main program. This allows generating complex experiment programs in a short time. All acquired data and reference values are time stamped using synchronised clocks of the Trigger Timer and Event (TTE) system. Diagnostics not yet fully integrated into the segment control system are synchronised by predefined triggers with TTE time stamps. This contribution will present the experiences with the flexibility of the W7-X Segment Control implementation.
P3.038

Application of the engineering standard for functional safety to the W7-X central safety system

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The commissioning and final validation of the central safety system and the acceptance by the authority were very important steps immediately before the successful ignition of the first plasma in Wendelstein 7-X in December 2016. Safety is the mandatory prerequisite for the operation of experimental devices of course to protect the personnel and the investment from hazardous situations. To fulfill these requirements the system has to control all dangerous situations with high availability and reliability. On the other hand the safety system has to be designed and realized very carefully to avoid safe but nuisance trips which might hinder experiments. Therefore the W7-X-team implemented a controlled process to develop and implement the Safety Integrated System (SIS) based on the international safety standard IEC 61511 (very similar to ANSI/ISA 84) for the process industry sector as guideline. This standard is based mainly on the concept of the safety lifecycle and safety integrity levels (SIL). The hazard and risk assessment delivers the allocation of the identified safety functions to different protection layers and the necessary SIL rating for the SIS. All activities to specify the requirements of the SIS, the design, the implementation, commissioning and validation follow the v-shape model proposed in the standard. Each single step is carried out with documented verification against the results of the previous development stages. In the end the validation shows whether the demands to the whole SIS are fulfilled. Of course the SIS must be operated and maintained in the future. Modifications and extensions in the future are foreseeable.
Model-based optimal scenario planning in EAST

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Ongoing work in the fusion community focuses on developing advanced plasma scenarios characterized by high plasma confinement, magnetohydrodynamic (MHD) stability, and noninductively driven plasma current. The toroidal current density profile, or alternatively the q profile, together with the normalized beta, are often used to characterize these advanced scenarios. The development of these advanced scenarios is experimentally carried out by specifying the device’s actuator trajectory waveforms, such as the total plasma current, the plasma density, the auxiliary heating and current-drive (H&CD) sources based on trial-and-error basis. In this work, a model-based numerical optimization algorithm is developed to complement the experimental effort of actuator trajectory planning in the EAST tokamak. The evolution of the q profile 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). In this work, the MDE is combined with physics-based correlations obtained from EAST experimental data for the plasma density, temperature, resistivity and non-inductive current drives to develop a control-oriented nonlinear PDE model. The optimization objective is to design feedforward trajectories for the plasma current, density, electron cyclotron heating, neutral beam injection and lower hybrid current drive that steer plasma to desired q profile and bN such that the achieved state is stationary in time. The optimization is subject to the plasma dynamics (described by the physics-based PDE model) and plasma state and actuator constraints, such as the maximum available amount of H&CD power and MHD stability limits. This defines a nonlinear, constrained optimization problem that is solved by employing sequential quadratic programming. The optimized actuator trajectories are assessed in nonlinear transport simulations in preparation for experimental tests in the EAST tokamak. This work has been supported by the US Department of Energy under DE-SC0010537.
Nonlinear burn control in tokamaks using In-vessel coils

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Control of the plasma density and temperature to produce a certain amount of fusion power, known as burn control, is one of the key issues that need to be solved for the success of tokamak fusion reactors such as ITER. In order to reach a high fusion power to auxiliary power ratio, tokamaks must operate near temperature and density stability limits. Therefore, active control to maintain a desired burn condition and avoid instabilities is absolutely necessary. Previous work makes use of mainly three different types of actuation: modulation of the auxiliary power, modulation of the fueling rate, and controlled injection of impurities. However, recent experiments showed the feasibility of modifying the plasma energy by using the in-vessel coils as actuators. Inspired by such experiments, a new burn control scheme is proposed in this work to exploit the in-vessel-coil system in combination with auxiliary power and fueling rate modulation. The in-vessel coils generate non-axisymmetric magnetic fields that modify the confinement of the plasma, which influences the plasma energy dynamics. By using the in-vessel coils, energy losses can be enhanced when needed and thermal excursions can be prevented. Moreover, actuation of the in-vessel coils may prevent the injection of impurities and its associated drawbacks. A control-oriented model has been developed to account for the influence of the in-vessel-coil currents on the plasma burn. While much previous work uses linearization techniques, a model-based nonlinear burn controller is proposed in this work. This nonlinear control approach is applicable to a larger range of operating conditions and is stable against a larger set of perturbations when compared with linear control approaches. The effectiveness of the controller is demonstrated via nonlinear simulation studies for different plasma scenarios. This work has been supported by the US Department of Energy under DE-SC0010661.
Nonlinear sliding mode control of the current density profile in tokamaks

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Research on fusion plasmas in tokamaks has led to the insight that the poloidal magnetic-flux distribution within the plasma has a crucial impact on its performance. Achieving certain types of poloidal magnetic-flux profiles, or alternatively certain types of q profiles, leads to resilience against undesirable instabilities and to higher bootstrap-current fractions, which in turn favor steady-state operation. To reliably and repeatedly achieve a desired q profile, feedback control is needed. Extensive work has recently gone on towards the development of q-profile feedback controllers. The nonlinearity of the plasma and the coupling between magnetic and kinetic variables demand a model-based control approach based on the magnetic-flux diffusion equation (MDE). The MDE is a nonlinear partial differential equation (PDE) modeling the time evolution of the poloidal magnetic-flux profile, and therefore of the q profile. Due to the complexity of the MDE, much of the previous work in this area used a linearized version of it for control design. While linear control approaches proved themselves effective in experiments, there is potential for improved performance by avoiding linearization and using the knowledge embedded in the nonlinear model to its fullest extent. One of the challenges associated with the design of model-based nonlinear q-profile feedback controllers arises from the fact that the model is non-affine in control, i.e. the q-profile dynamics depend nonlinearly on the control inputs (e.g., total plasma current and H&CD powers). In this work, we develop and test in simulations a nonlinear sliding mode controller for q-profile regulation that takes into account all the nonlinearities of the model. Assessment of the robustness of the proposed controller against unmodeled dynamics and perturbations, which is in general an advantageous characteristic of sliding mode controllers, is also part of this work. This work has been supported by the US Department of Energy under DE-SC0010661.
Physics-based control-oriented modeling of the current density profile evolution in NSTX-Upgrade

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Active control of the toroidal current density profile is among those plasma control milestones that the National Spherical Tokamak eXperiment - Upgrade (NSTX-U) program must achieve to realize its next-step operational goals characterized by the high-performance, MHD-stable plasma operation with neutral beam heating, and longer pulse durations. Motivated by the coupled, nonlinear, multivariable, distributed-parameter plasma dynamics, the first step towards feedback control design is the development of a physics-based, control-oriented model for the current profile evolution in response to non-inductive current drives and heating systems. The evolution of the toroidal current density profile 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 proposed control-oriented model predicts the spatial-temporal evolution of the current density profile by combining the nonlinear MDE with physics-based correlations obtained at NSTX-U for the electron density, electron temperature, and non-inductive current drives (neutral beams). The resulting first-principles-driven, control-oriented model is tailored for NSTX-U based on the predictions of the time-dependent transport code TRANSP. Main objectives and possible challenges associated with the use of the developed model for the design of both feedforward and feedback controllers are also discussed. This work has been supported by the U.S. Department of Energy under contract number DE-AC02-09CH11466.
In this work we present a new real-time acquisition and elaboration system for the two-color scanning beam interferometer installed on FTU. The real-time system provides the density informations that can be used to approximate the plasma and runaway beam radial position. Furthermore, the central chord plasma line density will be used to substitute the actual feedback signal for the fueling controller, that runs on the main real-time feedback control system. The system architecture is based on MARTe framework, running under a Linux operation system installed on a industrial controller, tuned for this application. For the acquisition of interferometric data (10 channels at 1.5 MHz), we adopt three high speed acquisition boards and one Reflective Memory (RFM) module to share data between nodes of our real-time network. The three boards are externally synchronized by mean of 30 MHz clock and gate signals. The first two DAQ boards have been devoted to the acquisition of 4 channels: sen(theta), cos(theta) (where theta is the phase) from each CO2 and CO lasers beam. The third board is dedicated to the data acquisition of the scanning system (CRS: counter rotating system) that moves backwards and forwards each probe beam with a 8kHz frequency. Each millisecond the system collects 1500 samples from each channel and reads the plasma current using the RFM. After the acquisition step, the software corrects the sen(theta), cos(theta) and CRS signals removing the offset from the two probing beam laser and scanning system. Then the phase of the CO2 and CO probing laser beams are evaluated and the total line density is computed with an average over 1 ms. Finally, using CRS information, the new system splits the total line density into 32 vertical chords with different major radii, and distributes them over the shared memory network.
Basic concepts and implementation strategy of the plasma discharge command sequencer for FTU Tokamak

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The plasma pulse phase of Frascati Tokamak Upgrade (FTU) is driven by the dedicated system FSC (Fast Sequence Control), which has been developed in order to send all the necessary commands to the different power plants feeding the toroidal and poloidal coils during the plasma discharge, meanwhile controlling the correct outcome. In case of incorrect execution of the sequence the system is able to emit an alternative recovery sequence in order to safely shutdown the power plants and the plasma pulse. Furthermore the FSC has an embedded event recording feature to record and timestamp all the commands, states and possible alarms. The FSC system has been recently upgraded porting it to an up to date off the shelf hardware platform, taking this opportunity to revise and update the basic concepts of the system and increase its flexibility in view of applications to future fusion machines. In detail the system is equipped with an integrated configurator in order to set it for the different sub-plants using modular concepts. A specially developed sequence programming language allows to program the sequences referring to logical names, as defined in configuration. The logical variables representing the global times that are stored in the real-time database of the FTU supervisory control system can be used in the sequence to relate it to the plasma discharge to be executed. The sequence programming language provides an abstraction level allowing the user to symbolically represent the actuation times of the different commands and the state verifications that can be linked in different ways. During the plasma discharge preparatory phase the global time parameters are transferred to the FSC control system, which performs a compilation of the sequence generating the absolute times and the binary files representing the full sequence; these are then loaded and implemented by the FSC Hardware.
Development of MPPC-based detectors for high count rate
DT campaigns at JET

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The products of fusion reactions at JET are measured using different diagnostic techniques. One of the methods is based on measurements of gamma-rays, originating from reactions between fast ions and plasma impurities. During the forthcoming deuterium-tritium (DT) campaign a particular attention will be paid to 4.44 MeV gamma-rays emitted in the $^{9}\text{Be}(\alpha,n\gamma)^{12}\text{C}$ reaction. Gamma-ray detectors foreseen for measurements in DT campaigns have to register spectra at high count rates, up to approximately 1 MHz. For the Gamma-ray Camera at JET a new setup will be based on a CeBr₃ or LaBr₃ scintillator and a multi-pixel photon counter (MPPC). We present two methods of shortening output signals in modules based on MPPC. A short detector output signal is necessary in order to minimize the number of pile up events at high count rates. One method uses a passive RC circuit with a pole zero cancellation, whereas an active transimpedance amplifier is used in the other one. The stability of the peak position and energy resolution as a function of counting rate was measured with a $^{137}\text{Cs}$ source. Due to the strong dependence of MPPC properties on temperature variation, a special device MTCD@NCBJ was designed and produced to stabilize the gain in MPPC-based scintillation detectors. We show that this device guarantees stable working conditions. This scientific work was partly supported by Polish Ministry of Science and Higher Education within the framework of the scientific financial resources in the years 2015-2017 allocated for the realization of the international co-financed project. 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.
Upgrade of the tangential gamma-ray spectrometer beam-line for JET DT experiments

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The JET tangential gamma-ray spectrometer (KM6T) is undergoing an extensive upgrade in order to make it compatible with the forthcoming deuterium-tritium (DT) experiments. The paper will present the design of the main components for the upgrade of the spectrometer beam-line: tandem collimators, gamma-ray shields, and neutron attenuators. The existing KM6T tandem collimators will be upgraded by installing two additional collimator modules. Two gamma-ray shields will define the gamma-ray Field-of-View at the detector end of the spectrometer Line-of-Sight. A set of three lithium hydride neutron attenuators will be used to control the level of the fast neutron flux on the gamma-ray detectors. The design of a combined movable gamma-ray shield and neutron attenuator will provide a choice of three operational conditions for deuterium and DT experiments, including that of a gamma-ray shutter. In addition to the beam-line upgrade, other components of the KM6T spectrometer are being upgraded. The present BGO detector is going to be replaced by two detectors, based on the LaBr₃ and CeBr₃ scintillators. The new scintillator detectors will provide a significant improvement in terms of energy resolution (<3% at 662keV) and count rate capability (up to 1 MHz). The existing obsolete data acquisition will be replaced by a high performance FPGA-based digital data acquisition system. The results of a feasibility study for the replacement of the radiation shielding components towards the JET diagnostics hall will also be presented. The study was done following one of the initial operational requirements of the diagnostics upgrade which was to provide means for frequent swapping of the two new gamma-ray detectors. The design of the upgraded spectrometer beam-line has been supported by extensive radiation (neutron and photon) transport calculations using both large volume and point radiation sources. The numerical results of these calculations will be also presented in the paper.
CeBr\textsubscript{3}–based detector for Gamma Spectrometer Upgrade at JET

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The diagnostic of fast ions at JET is based on the measurements of gamma-rays which are produced as a result of nuclear reactions between ions and plasma impurities. The gamma-ray spectra provide information on energetic tail of ion energy distribution. The existent BGO detector, with a decay time of \~300 ns, is sufficient during DD campaigns. The strong neutron and gamma-ray fluxes during D-T experiments induces new requirements for the detector. In addition to good energy resolution it must also be characterized by high signal-to-noise ratio and allow to perform measurements at high counting rate. The scintillators which fulfill these requirements are, among others, LaBr\textsubscript{3}:Ce (already tested at JET) and CeBr\textsubscript{3} with a decay time of \~20 ns. We report on measurements performed with a detector module equipped with a CeBr\textsubscript{3} scintillator. The measurements were made with standard gamma-ray sources. A strong \textsuperscript{137}\textsuperscript{137}Cs source, with an activity about 400 MBq, was used to provide high counting rates. The spectra were registered with a standard voltage divider and then with an active voltage divider, specially designed at NCBJ for high counting rates. The comparison of measured and Monte Carlo simulated spectra will be presented. This scientific work was partly supported by Polish Ministry of Science and Higher Education within the framework of the scientific financial resources in the years 2015-2017 allocated for the realization of the international co-financed project. 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.
Conceptual design of a gamma-ray monitor for lost alpha particles in JET

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A new diagnostics technique, the Lost Alpha Monitor (LAM), for the investigation of escaping alpha particles in JET has been proposed [1]. The method is based on the detection of the gamma radiation induced by the escaping particles on a target external to the plasma. For a beryllium target this reaction is $^{9}$Be($a$, $n\gamma$)$^{12}$C. The implementation on JET of the LAM technique would make possible correlated measurements of lost and confined alphas using the same nuclear reaction. The paper presents the conceptual design of the LAM diagnostics for JET. The main components of the proposed LAM diagnostics include a radiation collimator and shield which houses two gamma-ray detectors located behind lithium hydride neutron attenuators. The radiation shield is made up of a core stainless-steel collimator surrounded by the neutron and gamma-ray shield constructed from thick plates of high density polyethylene and lead. The collimator-shield assembly is placed behind the existing KJ5 soft X-ray camera in octant 4. The KJ5 soft X-ray camera shield is used as a pre-collimator for the LAM diagnostics. The fields-of-view of the LAM gamma-ray detectors are actually defined by the KJ5 collimator. Two solutions have been considered for the LAM beryllium target. The first proposal is to extend one of the TAE antenna protection tiles, while the other considers a separate dedicated target. The LAM gamma-ray detectors are based on the CeBr3 scintillators [2] coupled to metal channel dynode photomultipliers. The solution for the data acquisition is based on the ATCA data acquisition platform which includes fast digitizers [3].

Overview of property degradation of metallic mirrors for diagnostics in current and future reactors

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All optical spectroscopy and imaging diagnostics in next-step fusion devices will be based on metallic mirrors. The performance of mirrors is studied in present-day tokamaks and under laboratory conditions. This work deals with comprehensive tests of mirrors: (i) exposed in JET with the ITER-Like Wall (JET-ILW); (b) irradiation by He and heavy ions to simulate the impact of neutrons under reactor conditions. First Mirror Test at JET for ITER. Mo mirrors placed in pan-pipe cassettes were exposed to plasma in the main chamber wall and in the divertor inner, outer and base. In the main chamber only mirrors located at the entrance to the cassette lost reflectivity (Be deposition from the eroded limiters), while those in the channels were slightly affected. The performance of mirrors in the JET-ILW divertor was strongly degraded by deposition of beryllium, tungsten and other species. Splashing of metal droplets on mirrors also occurred. It should be stressed, that these solid Mo mirrors were not damaged by arcing. Radiation damage in mirrors: work towards DEMO. Optically active layer in mirrors is 20-30 nm thick. The conditions for the ion irradiation, He$^+$ (1-2 keV) and $^{98}$Mo$^{1+}$ (30 keV), were based on SRIM simulations. Studies were performed for mirrors irradiated by a single type and by both types of ions. The stepwise irradiation up to 30 dpa by $^{98}$Mo$^{1+}$ caused only small changes in the optical performance. Much stronger effects have been produced by helium because of bubble formation which led to the reflectivity decrease by more than 20%. Helium retention studies revealed that only 9% of the implanted He was retained. Consequences of various plasma-wall interaction processes on the performance and reliability of plasma diagnostic systems will be discussed.
Physical characterization of the JET operational space regarding EMLs by means of discriminant analysis

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High performance H-mode plasmas are characterized by short, repetitive edge perturbations known as edge-localized modes (ELMs). Large, unmitigated ELMs can result in significant transient heat loads released onto the plasma-facing components. Hence, characterization of ELMs and their control are crucial for avoiding a significant reduction in the divertor lifetime. This necessitates discrimination of different observed classes of ELMs and determination of operational boundaries for ELMy regimes. In this work, a parametric statistical classification system for ELM types using discriminant analysis has been developed and has been applied for the classification of type I and type III ELMs in a set of JET carbon wall plasmas. The classifier provides success rates up to 90% and renders itself as a fast, standardized classifier of ELM types, complementary to phenomenological approaches based on human expertise. Further, linear discriminant functions are constructed for determining the boundary between type I and type III ELMy regimes, both in terms of plasma engineering parameters as well as dimensionless physics parameters. The functions provide an insight into the dependence of the boundary on the plasma and machine conditions and identify the parameters which contribute most to the type I/III boundary. The classifier for ELM behavior developed in this work significantly reduces the effort of ELM experts in identifying ELM types, while the boundary in terms of engineering or physics parameters provides insight into the range of conditions under which specific ELM behavior occurs.
Engineering design and analysis of an ITER-like first mirror test assembly on JET

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The ITER first mirrors are the components of optical diagnostic systems closest to the plasma. Deposition may build up on the surfaces of the mirror affecting their ability to fulfil their function. However, physics modelling of this layer growth is fraught with uncertainty. A new experiment is underway on JET, under contract to ITER, with primary objective to test if, under realistic plasma and wall material conditions and with ITER-like first mirror aperture geometry, deposits do grow on first mirrors. This paper describes the engineering design and analysis of this mirror test assembly. The layout of the mirror assembly is dictated by the ITER requirements, with the directions of the aperture cones and the pinhole sizes are chosen to be ITER relevant. The only available in-vessel supports for this assembly are welded mounting brackets no longer used by other deposition/erosion diagnostics. Tests on mock-ups and calculations define the maximum load these brackets can take. The mirrors are very close to the plasma resulting in conflicting electromagnetic and thermal requirements. The components need to be sufficiently massive to cope with the thermal loads (setting a minimum wall thickness), but at the same time resistive enough to keep the disruption loads within those allowed by the mounting brackets. In addition, installation must be performed fully by Remote Handling only. The design evolved into a three part structure: interface — support — housing-aperture cones. Wall thicknesses were minimized, the housing surfaces are plasma sprayed with alumina and the support shape was also designed minimizing the formation of current loops. The most challenging components to manufacture were the multi-cone apertures. This was not suitable for conventional machining, hence additive manufacturing was used. The assembly was installed in the 2014-5 shutdown and will be removed in the 2016-17 shutdown.
Remote handling connector development for the ITER divertor diagnostics

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ITER fusion reactor is a very complex machine which has several different subsystems. It is still a research reactor and the testing results will be implemented in the next generation reactors. In the testing phase of the reactor there will be several sensors and instruments assembled inside the vessel for diagnostics purposes. One of the key diagnostics areas will be the divertor environment. Due to replacement process there must be connectors between the cables inside divertor cassette and vacuum vessel. The connectors should be remotely accessible. In the vacuum vessel there will be 17 divertors that will have diagnostics attached with electric wires. The amount of those wires varies between 30 and 202 in the cassettes. Total quantity of wires is over 2500, so the average of wires in each cassette is about 125. Some of the instrumented cassettes will be in standard cassettes and some central cassette depending to their location to the ports. Both standard cassette and central cassette can have different connector solutions. In addition the left and right cassette around the central cassette can have separate solutions depending on the space available and the structure of the connector. All the connectors must be connected by remote handling system because of the harsh environment inside the vessel. In a development project seven concepts for the connectors were developed and analysed. The concepts were created by 3D modelling and simulation. Some critical parts and functionalities were analysed and tested with mock-ups. The selection criteria were created according to the requirements set for the connectors. An evaluation matrix was used for the analysis and expert evaluation was implemented with scaled factors. From the analysis three solutions were selected for the further development. In this paper the concept creation, the analysis and further research topics will be presented.
System engineering challenges of Tokamak Services for Diagnostics consortium

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Electrical Services provide the electrical infrastructure to serve the diagnostics installed on the ITER Tokamak. The components of the 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. These electrical components (cables, connectors, feedthroughs, looms, etc.) are under the scope of the Electrical Services. All of the components must comply with the requirements of the ITER Vacuum Handbook, Codes and Standards, Electrical Handbook, etc. This paper outlines the System Engineering challenges of Tokamak Services for Diagnostics consortium (TSD). During the last 4 years under an F4E contract, the TSD group has carried out several tasks in four areas, in close collaboration with F4E and IO team members. These are the creation of Requirements, which are a documented representation of a capability or property that a system shall have; producing the Schematics (SSD), which will be used to follow signal path from sensor (start) to Diagnostic Hall (end); implementation of space reservation models for several diagnostic components into the IO ENOVIA Database and creating/updating the Interface documents, which are the common boundaries between co-functioning Systems. The main purpose of TSD is to prepare the technical documentation for the Feedthroughs, Looms and for the RH Connectors. These documents are the ones which will be given to the industrial companies who will manufacture the components. During this period TSD had several achievements on creation of CAD models (developed feedthrough, tail, and junction box space reservations), on the Schematics (development of Upper and Lower port SSDs), on the Requirements (around 4000 collected requirements) and on the Interfaces (40 IS has been created/updated).
Numerous plasma-near mirrors of optical diagnostics of ITER require protection from erosion and deposition caused by impinging energetic particles. This is achieved by approximately 60 individual Diagnostic Shutters, rather simple mechanical devices which obstruct the mirror’s sight towards the plasma when the diagnostic is not in use. If a shutter fails to operate, so does the respective diagnostic. Shutters shall operate in vacuum under high thermal fluxes over 20 years without maintenance. Their components will experience neutron fluxes of up to $10^{14}$ $1/cm^2s$ with energies up to 14 MeV. As these conditions are unprecedented even on fusion devices, standard engineering solutions are ruled out, and qualification R&D efforts are extensive. As every shutter is part of the diagnostic it protects, the design tasks are widespread among Domestic Agencies and their suppliers. However, the obvious synergy potential of these highly resembling systems calls for coordination of design and prototyping to save effort, but also consistently handle risk. Therefore, a coordination strategy for all ITER shutters was implemented at IO. An extensive collection of experience on shutters from 14 fusion devices was performed, including failure reports. This experience is summarized in the present work. For the first time, the state-of-the-art of shutter design with respect to fusion diagnostics is thereby defined. The lessons learnt are assessed with respect to their applicability for ITER. Furthermore, potentially design-driving environmental effects such as high-temperature creep and irradiation-induced embrittlement are recalled and theoretically evaluated against the specific ITER operational conditions. The findings of both assessments are put into context with the current designs of all ITER shutters. In a next step, these are reviewed with particular emphasis on possible synergies between different shutter systems. Finally, recommendations on design and necessary R&D, such as common prototyping and the development of generic components are given.
Can gas puffs be used for cleaning of the diagnostic first mirrors in ITER?

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First mirrors are plasma facing components which redirect light to the protected optical diagnostics. Initial investigations [A. Litnovsky et al. Nuclear Fusion 49 (2009) 075015, V. Kotov et al. Fusion Eng. Des. 89 (2011) 1583] showed that deposition of impurities (Be, Fe etc.) may cause drastic degradation of the mirror reflectivity and thus severely restrict the diagnostic performance. Very moderate loss of reflectivity was found experimentally under net erosion conditions on the mirror [A. Litnovsky, Fusion Eng. Des. 82 (2007) 123], which would be therefore preferable from the point of view of the diagnostic performance. It is planned that the diagnostic ducts in the ITER main chamber will be housed in the recessed port plugs which have no direct contact with plasma. Erosion in those areas is expected mainly due to high energetic (charge-exchange) atoms. Local gas puff of deuterium (D2) in front of the port plug can create large fluxes of such atoms and thus enhance re-erosion of deposits. At the same time, erosion from the surfaces in the vicinity of the diagnostic entrance aperture can lead to increased incident fluxes of impurities and increased deposition rate. Hence, accurate numerical calculations are required to estimate the net effect (erosion minus deposition). In the present paper the transport of neutral particles is modeled with the kinetic Monte-Carlo code EIRENE taking into account 3D geometry of the first wall and ports. The plasma parameters are prescribed based on available experimental data and B2-EIRENE (SOLPS) simulations of the ITER edge plasma. Preliminary studies show that in the vicinity of gas inlet on the port-plug surface the calculated erosion rate of Be is at least a factor of ten larger than its deposition rate. That is, it is highly probable that the gas puff can shift the balance towards net erosion.
Rigid and adjustable fixation of the shielding modules into the ITER port plugs

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During the final design review of Diagnostic Port Plugs, it has been highlighted that the current system of fixation, based on gaps, while it is not harmful for the port plug, it throws large uncertainties over the alignment of the optical systems placed inside the DSMs at the same time that the real mechanical behaviour of the assembly is clearly unknown. Due to the fact that the DSM is not rigidly fixed to the Port Plug structure, the DSM and its internal components suffer, during abnormal events like seismic or electromagnetic disruptions, an important level of amplification on the accelerations induced. This magnification is largely derived from a rattling effect which the DSM is vibrating inside the port plug. This effect produces an increase of the efforts acting on the anchoring components of the port plug at the same time that the final position of the DSM after an abnormal event is unknown and greatly influenced by the manufacturing and assembly tolerances. In order to solve all these issues, a new locking system has been developed based on a rigid configuration. Non relevant modifications are required in the Port Plug structure or in the DSMs for the implementation of this new system because the interfaces are kept intact. Results based on this new configuration are available and they present an important attenuation of the induced accelerations on components inside the DSMs. In addition the system permits the individual adjustment of each one of the interfacing component to finally achieve the correct aligned positions. Therefore, this new locking system has been assumed for all IO Diagnostic Port Plugs.
P3.057

Overview of the Loom Electrical Vacuum Interface preliminary design in ITER Equatorial Port #17

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ITER will have a set of 45 diagnostics to ensure controlled operation. Many of them are integrated in the ITER ports. Housed in generic structures, this modular integration is designed to help diagnostics withstanding the plasma loads whilst complying with the French regulations. Now that the Domestic Agencies and ITER Organization are developing the preliminary or even final designs of the diagnostics on ITER, it is important to provide the designers with a flexible infrastructure to allow the efficient development of the diagnostic systems. Interface requirements have been defined for common features in the aim of sharing the designs in order to minimize the effort of developing very similar components whose qualification to ITER requirements is expensive and schedule constrained. Recent progress has been made in the definition of a generic concept of an electrical assembly ensuring in-vessel connections to the Diagnostic Shield Modules hosting the front end components. A remote handling compatible generic design of an electrical assembly that can be adapted for each of the Diagnostics equatorial and upper Port Plugs has been defined. The associated electrical feedthrough so called Loom Electrical Vacuum Interface (LEVI) has been designed addressing the severe constraints of high vacuum compatibility and compliance to nuclear safety rules and remote handling compatibility. A solution using mineral insulated cables brazed on a double confined stainless steel structure will be presented and discussed. A particular port, the equatorial port 17, is used to illustrate the concept and the main challenges to overcome.
P3.058

Development of Load Specifications for the Design of ITER Diagnostic System and Port Integration

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ITER is the world’s largest fusion device currently under construction in the South of France with over 60 diagnostic systems to be installed inside the port plugs, the interspace or the port cell region of various diagnostic ports. The plasma facing Diagnostic First Wall (DFW) and its supporting Diagnostic Shielding Modules (DSM) are designed to protect front-end diagnostics from plasma neutron and radiation while providing apertures for diagnostic access to the plasma. The design of ITER port plug structures (PPS) including the DFWs and DSMs is largely driven by the electromagnetic loads induced on these passive structural components during plasma disruptions and vertical displacement events (VDEs). Unlike the DFW and DSM, the design of ITER diagnostic system, however, is likely driven by the steady-state thermal loads from plasma volumetric and surface heating; and the dynamic response of in-port components attached to the port-specific DSM or PPS as a result of transient electromagnetic loads induced on the Vacuum Vessel (VV) and the port extension during asymmetric plasma VDEs. We investigate in this study the worst plasma disruption load scenarios for diagnostic systems of varying size and location, and summarize the steady-state thermal, transient electromagnetic and VV VDE inertial loads for equatorial port and upper port diagnostic systems such as toroidal interferometer and polarimeter (TIP), electron cyclotron emission (ECE) in E9 and wide angle viewing (WAV) system in U14 as a result of VV movements during asymmetric VDEs. We also introduce the common approach and design requirement for tenant interface load transfer for in-port diagnostic systems attached to the DSMs as one of the design engineering and integration tasks of the ITER diagnostic ports.
To achieve the real time controllability of plasma, real-time network is required in fusion experiments place. KSTAR Plasma control system (PCS) adopted the reflective memory (RFM) as a real time network. Since RFM based network has low latency and low jitter. However, KSTAR is also adopted Synchronous Data bus Network (SDN) as real time network to provide real time network to fueling system. Since the performance of SDN is comparable to RFM and SDN can be implemented by existing 10Gb Ethernet network. SDN is ITER real time network based on UDP multicast atop 10GbE cut-through packet switching infrastructure. To provide the connectivity of between these real-time network, we developed the translator between RFM and SDN. The translator read data from RFM, make predefined SDN packet, and send packet to destination in real time. Moreover, the translator receive SDN packet and write data to predefined data address in real time. This translator translate each data in accordance with KSTAR timing system. Therefore, each data synchronize with KSTAR timing system. Moreover, the translator also provide the monitoring functionality of SDN and RFM using EPICS (Experimental Physics and Industrial Control System). This paper presents the algorithm of the translator system and experiment setup. The translator was tested on the MRG-R kernel 3.10. This experiments show a real time performance of the translator.
Implementing a Neutron-Diagnostic use case in a PXIe platform through a 3D remote laboratory

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The iRIO-3DLab platform has been devised to enhance the learning process and reduce the development time for engineers in charge of designing intelligent DAQ systems based on PXIe technology and distributed control systems such as EPICS. iRIO-3DLab consists of an Opensim-based virtual world that aims to promote the understanding of how such a kind of DAQ system works, and how the EPICS IOC should be configured according to the configuration and monitoring demanded by diagnostic requirements. This contribution describes how to use iRIO-3DLab to implement a Neutron-Diagnostic use case using a PXIe platform, which includes: setting up the hardware remotely and configuring the EPICS-IOC and the interface with the application using CSS. The PXIe platform includes: PICMG 1.3 compliant computer, PXIe chassis, PXI synchronization module using IEEE 1588 standard, PXIe FlexRIO FPGA-based device, and adapter module with four 14-bit, 250 MS/s analog inputs. iRIO-3DLab allows users to connect the virtual replicas of this equipment to the cubicle and select the application to be implemented in the FPGA. The FPGA deals with pulse, campbelling and current mode measurements of the acquired signals. When the user selects the application there is an associated EPICS IOC implementing the data gathering. For each action performed by the user a message is sent to the real system through the Internet, so the real IOC matches this virtual configuration. The virtual world allows the user, firstly to set the waveform generator used to generate predefined signal patterns in order to check the correct measurement, and secondly to interface with the OPI panel to assess the results obtained. Finally, the contribution describes the correlation of the actions executed by the user and the internal command and data flow of the hardware and software platform to understand all the technologies involved in this complex DAQ setups.
P3.062

Integration of EPICS based monitoring for Ion Cyclotron High Voltage Power Supply

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A Dual output (27kV & 15kV), 3MW High Voltage Power Supply (ICHVPS) has been installed and integrated with a Diacrode based RF source to be used for ICRF system. The ICHVPS Controller is based on LabVIEW Real-time PXI controller, which supports all control and monitoring operations of the PSM based power supply. The controller supports all essential features like, fast dynamics, low ripple and protection for source and loads. EPICS (Experimental Physics and Industrial Control Systems) is an open source software widely accepted in scientific community as supervisory distributed control system including CODAC (Control, Data Access and Communication) for ITER. However interface with LabVIEW RT systems is not fully matured yet. Evolving requirements of interface with platforms like EPICS with ICHVPS control has been assessed and implemented for monitoring purpose. A test case was implemented to identify compatibility, feasibility, consistency and performance of EPICS Input Output Controller (IOCs) server implemented in Real-time controller. This paper discusses integration of EPICS IOCs and LabVIEW based Real-time Controller along with some analysis on limitations imposed by such integration.
EPICS device support for ATCA Board with hotplug capabilities

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The Advanced Telecommunications Computing Architecture (ATCA) standard defines a high performance technical solution that meets the requirements for fast controllers on large-scale physics experiments like ITER. This platform provides high throughput, scalability and features for high availability such as redundancy and intelligent platform management which are essential for steady state experiments. An ATCA Control and Data Acquisition (CDAQ) Board was developed for the ITER Fast Plant System Controller (FPSC) project, which is already available in the ITER Catalog. This board comprises 48 galvanic isolated analog and/or digital channels configurable as input or output with digital signal processing capabilities performed by a Field Programmable Gate Array (FPGA). The Enhanced Physics and Industrial Control System (EPICS) is a set of open source software tools, libraries and applications used worldwide to create distributed soft real-time control systems for scientific instruments. To provide the hardware integration in the EPICS environment, a device support has been developed as a software layer, which is comparable to the abstraction layer provided by the device driver on the Operating Systems. This paper presents the implementation and test of an EPICS Device Support for the ATCA CDAQ Board which provides templates for easy configuration of the entire system. This solution also allows simultaneous and independent acquisition by each board, providing hotplug features which support insertion and removal of boards while keeping other modules and the overall system running. Operation with several boards and different versions of Linux operating system was performed and the results are presented.
Taking advantage of intercommunication features of IPMCs to implement PCIe hot-plug in ATCA systems

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Control and Data Acquisition (CDAQ) systems applied to large physics experiments like ITER, are designed, among other features, for High-Availability (HA). A CDAQ system based on the PCI Industrial Computer Manufacturers Group (PICMG) 3.x AdvancedTCA Base Specification and Intelligent Platform Management Interface (IPMI) standards grants these features. One of the key functions of the HA is the hot swapping possibility of the CDAQ system elements which is granted by those standards. An Advanced Telecommunications Computers Architecture (ATCA) CDAQ system can be designed implementing the Peripheral Component Interconnect Express (PCIe) standard communication protocol in their fabric lines. However this protocol has its own rules for hot swapping, defined as hot plug, ATCA and PCIe standards are not completely compatible about this subject. Instituto de Plasmas e Fusão Nuclear (IPFN) has developed and implemented a CDAQ system using the ATCA and PCIe standards. This paper describes the software architecture and functions implemented in the microcontroller of the Intelligent Platform Management Controller (IPMC) of the ATCA boards in order to bring those standards compatible in what concerns hot-swap and hot-plug processes. The result was mainly achieved taking advantage of the intercommunication features of IPMCs through the Intelligent Platform Management Bus-0 (IPMB-0).
PCle hot-plug event handling tasks using PICMG standard interrupt mechanism for ATCA based instrumentation

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The Advanced Telecommunications Computing Architecture (ATCA) specification implements important key features such as high reliability, high availability, redundancy and serviceability for control and data acquisition instrumentation fault condition, hardware malfunction, firmware updates and hardware reconfiguration. Red Hat Enterprise Linux and corresponding kernels already have built-in mechanisms and embedded software for Peripheral Component Interconnect Express (PCle) hot-plug support that allows automatically remove of PCle device nodes and associated device files from the system providing a fast replacement strategy for damaged cards without require an entire system shutdown. This paper describes handling of PCle hot-plug events at a software middle level using the PCI Industrial Computer Manufacturers Group (PICMG) standard interrupt mechanism. The handling tasks can be accomplished by ATCA cards chipsets with support to PCle hot-plug features, Linux hot-plug embedded controller and Red Hat built-in device manager module. The goal is to implement a fast hardware replacement solution without system shutdown providing high availability capabilities to ATCA control and data acquisition instrumentation specially directed for large fusion experiments such as International Thermonuclear Experiment Reactor (ITER).
Real-Time data acquisition Prototype proposal of the ITER radial neutron camera and gamma-ray spectrometer

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The Radial Neutron Camera (RNC) and the Radial Gamma-Ray Spectrometer (RGRS) are two ITER diagnostics, devoted, respectively, to the real-time measurement of the neutron emissivity profile (to be used for plasma control purposes) and to the measurement of the confined alpha profile and runaway electrons. The two systems are closely related as they share the same equatorial port plug and part of the lines of sight and both require the acquisition of event-based signals from radiation detectors. The RNC Data Acquisition and Processing (DAQP) system should be capable of handling peak count rates of the order of \(10^6\) counts/s for a time duration up to 500 s. In order to identify and study critical issues, a DAQP prototype will be developed based on an Evaluation Board from Xilinx featuring a \(×8\) PCIe Gen3 interface and an Input/Output expansion connector which allows the insertion of a FPGA Mezzanine Card with 2 input channels of 12-bit sampling at 1.6GHz. The system is based on a computer capable of hosting \(2×8\) PCIe evaluation boards with 4 input channels allowing a data throughput bandwidth of up to 16GB/s from the digitizers to the host. The design activities for the DAQP of the RGRS diagnostic are limited to the proposal of a conceptual design interfacing correctly with the RNC, thus excluding any prototyping. As the two diagnostics have similar features in the long term, the RGRS DAQP design will benefit from the results of the RNC prototype tests. This paper will: a) present the RNC DAQP prototype showing its compliancy with the RNC plant system Fast Controller; b) show the scalability of the actual RNC DAQP from the prototype concept; c) describe the RGRS DAQP system and its interface to the ITER CODAC; d) identify the differences between the RNC and RGRS DAQP needs.
Embedded control systems for gyrotrons applications based on NI solutions

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The Swiss Plasma Center (SPC) is involved in the development and the operation of gyrotrons for fusion application (TCV tokamak, W7-X, ITER) and for medical application as well (spectroscopy DNP/NMR). In this framework, embedded control systems based on National Instrument (NI) compact Reconfigurable Input Output (cRIO) and compact Data AcQuisition (cDAQ) offer versatile solutions for dedicated applications. Three specific developments are presented and discussed here. First, a complete control system based on cDAQ material has been implemented in a Modulator Power Supply (MPS) controlling the anode to cathode voltage of a triode type gyrotron \cite{1}. This system placed at the cathode potential (84 kV) provide the MPS output voltage control, the system protections, an embedded data acquisition and the interface for the remote operation. The dynamic drive of the output voltage is made possible thanks to the FPGA integrated onto the cRIO chassis. Second, we developed a compact system to characterize the RF losses on the various gyrotron components. Based on a cDAQ solution, this system allows for real time calorimetric measurements on the water cooling circuits. Third, we developed a whole control system for a DNP/NMR gyrotron \cite{2}. Dynamic Nuclear Polarization (DNP) has emerged as a powerful technique to obtain signal-to-noise enhancements of a few orders of magnitude in nuclear magnetic resonance (NMR) spectra. The flexibility given by these compact control systems and acquisition could offer multiple solutions in many applications for fusion research.

\cite{1} U. Siravo et al., Fusion Engineering And Design, vol. 96-97, October 2015, p. 597-601
\cite{2} S. Alberti et al., Phys. Plasmas 19, 123102 (2012); doi: 10.1063/1.4769033
Data acquisition system for LN2 cool down experiment for 6 nos. of cryogenics pumps

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Cryogenic Instrumentation is a unique and vast field and requires an in-depth understanding of the process and instrumentation. 26 channels Data Acquisition System is required for the 6 nos. of Cryogenics Pumps LN2 cool down experiment. The data acquisition system measures 22 nos. of temperature signals, 2 nos. of level signals of the buffers and 2 nos. of Nitrogen Dewar Signals (Pressure and Level signals). The data for all the 26 channels is required to be monitored and acquired in the data acquisition system. This data will be helpful in analysis and understanding of the 6 pumps configuration along with many other issues. The channels sampling rate is kept 1Hz (1 sample/sec/channel). Slow acquisition is done for all the 26 channels. On the first day, the filing of Dewar was kept in series with the cool down of all the 6 pumps. So the cool down for the pumps took more time in comparison with the second day experiment. On second day the configuration was changed and cool down experiment was done which Dewar already filled upto 60%. The temperature signals and pressure and level signals (LN2 Dewar) acquired from the data acquisition system were needed in understanding the cool down rates for the pumps, to know the Dewar capability for 6 pumps and many other issues. The instrumentation, powering, data logger and the software for the 26 measurement channels will be discussed in details.
Commissioning and First Operation of the SPIDER Control and Data Acquisition System

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The Control and Data Acquisition System (CODAS) of SPIDER, the first experiment of the Neutral Beam Test Facility, is under installation and undergoing the commissioning and first operation phases. The system hardware is nearly compliant with the ITER CODAC catalog for slow and fast plant systems. The system software is based on a combination of software frameworks that altogether collaborate to provide the required system functions. Slow control is implemented through the ITER CODAC Core System that encompasses EPICS, Control System Studio and ITER specific tools such as the Self-Data Description and Maven. Data management relies on MDSplus for data acquisition, storage, access, and processing. Fast control is far beyond the time requirements that can be met by EPICS and, thus, is implemented through MARTe. The paper will provide an overview of the control functions realized in SPIDER CODAS along with a technical discussion on their implementation. The paper will then describe the problems encountered during the system commissioning and the proposed solutions. Finally the results of the commissioning and of the first operation will be reported.
A ‘Generic Tool’, adaptable to different configurations, to compute acting EM loads on Diagnostics

This paper is focused on the computation of EM loads induced by plasma current disruptions on the Diagnostics positioned inside the Equatorial Port Plugs, and more explicitly, on the creation of a detailed set of tools (Finite Element ‘FE’ models and routines) which allow the automatic characterization of the EM phenomena (DINA) as well as they provide versatility for the adding/removing of the different internal diagnostics. In this design phase of ITER, are many diagnostics whose level of maturity begins to be sufficiently elaborated to require a detailed study of the volumetric distribution of EM loads. For this specific aim, is desired a proper characterization of the electrical connections together with a correct definition of materials and geometries. The main goal of this investigation is the development of a ‘Generic Tool’ that can be used, for all internal systems, during the phases of design and analysis thanks its ability to adapt to changes in the design. In addition, not only components inside the DSMs need to be covered by EM analysis, exist some safety important components ‘SIC’ bolted or welded to the Port Plug Flange which needs to be assessed, therefore, in further design stages, this tool will be improved to cover in more accurate way all these systems.
Monte Carlo simulation of the neutron measurement for the Large Helical Device deuterium experiments

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The Large Helical Device (LHD) plans to start the deuterium experiment in March of 2017, where a maximum neutron yield of $2.1 \times 10^{16}$ neutrons/3 sec is expected. For the deuterium experiment, neutron flux monitors, a neutron profile monitor, a neutron activation system and other neutron detectors have been prepared. The characteristics of those neutron diagnostics, such as the detection efficiency, the plasma position sensitivity and the special resolution, have been evaluated by the neutron Monte Carlo simulation using a three-dimensional model of LHD, where the MCNP-6 Monte-Carlo neutronics code is used with the cross-section library of ENDF B-VI. In the MCNP calculation geometry, the LHD components within the helical coil support structure are divided by small toroidal angle pitch, and the components are assumed to be toroidally symmetric in a toroidal pitch angle. The geometry in one toroidal pitch angle is modeled based on the CAD drawing with some simplification. Three neutron flux monitors using a U-235 fission chamber with 50 mm-thick polyethylene moderator are located on the center axis of LHD and at two toroidal location on the horizontal plane outside the cryostat. It is confirmed that the detection efficiencies for total neutron emission of the LHD plasma are almost insensitive to the source neutron energy in the range of 2-3 MeV, the plasma position and the neutron source profile. The neutron profile monitor consists of 11 channels of stilbene detectors with a parallel collimator embedded in the concrete floor slab. It is found that the crosstalk in the adjoining channels is smaller than 1%.
Fuzzy logic based model of bolometer detector sensors and cameras

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Devices that are capable of measuring the total plasma radiation in fusion reactor experiments are indispensable for safe and reliable plasma operation. One of the most widespread type of these kind of devices are metal absorber–metal resistor bolometers where the radiation is absorbed by a metallic layer and the change of the layer’s temperature is measured by metal resistors. Based on the measured change of the resistance, the radiated power absorbed by the metallic layer is back-calculated. With this back-calculated power value the total plasma radiation (or spatial distribution thereof) can be deducted, once the geometrical properties of the observation (direction, solid angle, etc.) are known. In the process of the calculation of the plasma radiative power from the change of the resistance of a metallic meander, a number of assumptions have to be made the validity of which is normally hard to prove. To ascribe a given degree of trustability to the derived plasma radiative power, the use of a fuzzy logic based approach is herewith proposed and implemented, where fuzzy functions are assigned to different physical and geometrical properties of the metal resistor → metal absorber → camera housing → numerical assumptions back-calculation chain. This fuzzy inference system handles such absolutely independent factors like characteristics of the metallic layer (thickness, absorption coefficient, etc.) and the geometric setup of camera (tolerances in the line-of-sight, spatial position, etc.). Our fuzzy model is tested on results of numerical plasma simulations the uncertainties of which are also handled via fuzzy functions. The resulted fuzzy system (camera + plasma) is a suitable tool to be able to estimate the overall error and uncertainty of radiation detection. In case of actual bolometers, the model can be initialized with real data and then inserted (for example) into the plasma control system.
The stand-alone optimized post-processing algorithms for plasma diagnostics

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The development of GEM detector based acquisition systems resulted in the increase of throughput and resolution in the new revision of the system. The FPGA-based electronics is used to acquire, diagnose and to preliminarily analyze the data of soft X-ray emitted by hot plasma in Tokamak. Moreover, the development of electronics allowed to implement algorithms, so far performed offline after the experiment. Post-processing analysis consisted of calculating temporal, spatial and electric charge distribution in the detector. Due feasibility study with subsequent implementation were performed to assess the achievable throughput at which the data can be post-processed. The primary objective was to enhance the functionality of plasma detection systems and, subsequently, to enhance the capabilities of plasma control mechanisms with the introduced systems. The study concerns the implementation of stand-alone version of algorithms developed by the authors for the plasma diagnostics. The choice of hardware and overview of implementation issues is presented. Previous study concerning achieving speedup in MATLAB and overview of hardware of PCIe cards and processors continued, which led to justification of choosing optimal solution for a stand-alone application on PC responsible for on-line analysis of data with highest throughput. Implementation based on Intel Intel Xeon multicore CPU was further investigated. The resulting speedup is given as compared to previous MATLAB implementations. The optimal implementation is presented with conclusions for further research and for with presenting further steps which should be undertaken to achieve higher throughput.
P3.074

Influence of the geometry system on the effectiveness of polarimetric measurements in the thermonuclear plasma diagnostics

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On the basis of the angle variables technique (AVT) changes of polarimetry state of electromagnetic wave passing through the thermonuclear plasma in the poloidal plane have been analyzed. The first section analyzes the changes in polarization state depending on the angle at which the test beam was sent, for the same plasma parameters. Subsequently, for a given geometry, using numerical calculations the influence of changes in various parameters of plasma (plasma density, magnetic field, current density, the safety factor profile) on the state of polarization of the test beam have been verified. All numerical calculations were being performed for plasma parameters that occur in the real reactors. It should be emphasized that due to the different geometries in the present work both Faraday and Cotton-Mouton effect are taken into account.
P3.075

Preliminary assessment of ex-vessel waveguides for the ITER Plasma Position Reflectometer

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This work describes the preliminary assessment of the different waveguide technologies for the ex-vessel transmission lines of the Plasma Position Reflectometer (PPR) in ITER. Initially, both oversized rectangular and circular corrugated waveguides were considered for the study; the former due to reduced costs and ease of procurement and the latter due to better performance in terms of attenuation and radiation characteristics—which have a great impact in gaps and miter bends performance for instance-. Using the preliminary layout information available at this initial stage of the PPR design and the required frequency range, several waveguide dimensions were evaluated for both technologies, namely WR137, WR187, WR284, WR340 and WR430 in case of rectangular waveguides and 31.75 mm, 63.5 mm, 76.2 mm and 88.9 mm diameters for circular corrugated waveguides. These dimensions were chosen taking into account the space reservation provided by ITER and commercially available waveguide sizes. In an initial stage, only ohmic losses and mode conversion at mitre bends were taken into account for the comparison of all the possibilities. Results from a literature survey helped evaluating analytically the several possibilities in order to narrow the selection and set up some conclusions. In a later stage of the analysis, both ohmic loss and mode conversion at mitre bends were calculated with rigorous codes only for the selected geometries, together with mode conversion losses from gaps along the transmission line. This process led to the final estimation of performance for the preliminary routing of the PPR ex-vessel transmission lines. The reference solution of choice was 88.9 mm internal diameter circular corrugated waveguide, as the best compromise between in-band performance and space constraints.
Seismic Analysis of High-Power DC Reactor Prototype for ITER Poloidal Field Converter

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This paper mainly introduces the seismic analysis of the high-power dc reactor prototype, whose functions are to limit the ripple current and the increasing rate of fault current in the ITER poloidal field (PF) converter. The stacked reactors with the assembly dimension (L×W×H) of 2955 mm×1639 mm×3296 mm and weight about 5 tons are fixed to the steel base by five support components. In order to evaluate the seismic response of the structure under specific seismic excitation, a method based on response spectrum is adopted in this paper. The design earthquake spectrum of seismic level one (SL-1) with damping coefficient of 2% as well as 4% provided by ITER is applied as load. In addition, the simulation analysis is introduced in detail and the results as the displacements, stresses and reactive forces are also presented. This analysis could provide reference for the foundation design of dc reactor, as well as make a contribution to the similar seismic design work.
Free or confined arc model relevant to the quench hazard of large superconducting coils

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It is conceivable that electrical arcs can occur during the failure of a large superconducting magnet following an unmitigated quench accident. To assess such accidents, it is important to employ appropriate arc models to calculate the voltage current characteristics and heat dissipation as a function of conditions such as pressure and arc length. Although electrical arcs have been studied for many decades, the complex and destructive nature of arc phenomena has not allowed detailed models to be well established. During an unmitigated quench, resistive heating raises the conductor and the insulator temperature. Subsequently, the electrical and mechanical properties change. This can lead to dielectric breakdown of insulators and arc formation. Inline and bypass arcs can form that are sustained by the massive stored magnetic energy – of the order of gigajoules for ITER. If windings are bypassed by shorts, the arc current and the arc column diameter of inline arcs increases. Cable-in-conduit conductors limit the maximum arc column diameter and if limited, the arc properties change rapidly as the arc changes from a free arc to a confined arc. We assume ITER relevant conditions and for arc current 100 A – 100 kA we calculate the arc column electrical properties and temperature, by solving a set of equations describing the arc physics. The equations describe the arc column heating, gas ionisation, heat loss and electrical properties. By constraining the maximum arc column diameter in the solution, the transition between free and confined arc can be included. The calculations are compared to other relevant arc models and measurements to understand the sensitivity of their application to magnet safety assessment.
Series Production of ITER Toroidal Field Coil Double Pancakes in Japan

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Japan Atomic Energy Agency (JAEA), as Japan Domestic Agency, has responsibility to procure 9 ITER Toroidal Field (TF) coils. JAEA completed proto double-pancake (DP) trials aiming at qualification and optimization of manufacturing procedure of TF coil in 2015. Series production of DPs then started and winding of 14 DPs, heat treatment of 11 DPs, fabrication of 9 radial plates (RP), transfer of 7 DPs and cover plate (CP) welding of 4 DPs were completed until Feb. 2016. Challenging tight tolerances in conductor length, +/-0.01%, was achieved to transfer heat-treated conductor into RP groove. 1 mm flatness was achieved in RP, whose height and width are 13 m and 9 m. In addition, about 2 mm flatness was achieved after CP welding by optimizing welding sequence. The first DP insulation was completed and preparation of DP impregnation is underway. It is important to achieve good flatness, such as 2 mm, after impregnation of DP in order to simplify stacking process of DPs to form winding pack (WP). In the proto DP impregnation, about 2 mm flatness was successfully achieved by compensating out-of-plane deformation of DP after CP welding.
P3.080

Performance analysis of ITER CSI coil and conductor 1A of the ITER CSMC

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The ITER Central Solenoid Model Coil (CSMC) is a superconducting solenoid operated at the JAEA centre of Naka, Japan, since 2000 to test the performance of insert coils in its bore, where it produces a magnetic field of 13 T representative of the ITER CS operating conditions. In 2015, the ITER Central Solenoid Insert (CSI), whose Nb3Sn cable-in-conduit conductor (CICC) will be adopted for the 3L module of the ITER CS, was successfully tested in the bore of the CSMC. The test of the CSI performance allowed assessing its current sharing temperature after up to 16 thousand cycles in different operating conditions (in terms of magnetic field and current) corresponding to different phases of the plasma pulse, namely initial magnetization and end of burn. As a full-size short sample of the same CICC was previously tested in the SULTAN facility, SULTAN-like operating conditions were also reproduced to allow a comparison, with special reference to the effect of the Hoop strain, not present in the SULTAN straight sample tests. The data collected during the recent CSI tests at different cycles are reported in detail and, when applicable, compared to the SULTAN results. The 4C code is used to analyse the CSI performance. First the calibration of the free parameters of the model (the effective n-value and the extra longitudinal strain) is performed based on the measured voltage and jacket temperature Tjk along the CSI. Then 4C is used to compute the evolution of the conductor strand temperature, which is compared to the measured Tjk. The performance of the conductor 1A of the CSMC, also measured, are also analysed with the 4C code and the results are compared with the measurements performed in the previous test campaigns (2000, 2001, 2002 and 2008).
The ITER Central Solenoid Module final test facility*

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General Atomics (GA) is currently manufacturing the ITER Central Solenoid Modules (CSM) under contract to US ITER at Oak Ridge National Laboratory, under the sponsorship of the Department of Energy Office of Science. The contract includes the design and qualification of manufacturing processes and tooling necessary to fabricate seven CSM (6 + 1 spare) that constitute the ITER Central Solenoid. The modules will be produced and delivered to the ITER site during 2018–2020. Each CSM will undergo final testing at GA to verify performance. Testing includes helium leak testing, high voltage insulation testing, cooldown to 4.7K and charging to 48.5 kA followed by a series of tests designed to measure as-built performance of the superconductor. GA has completed the design and is now installing and commissioning the Final Test Facility at the CSM Manufacturing site in Poway, California. The facility includes a number of critical subsystems. The test chamber system consists of a 160m³ liquid nitrogen shielded cryostat to support the 110 ton CSM, a vacuum pumping and leak detection system and Paschen testing equipment. The feeder system connects the test chamber to the cryogenic and electrical systems. It includes a coil termination box (CTB), high temperature superconducting current leads and a superconducting feeder duct. The cryosystem provides refrigeration and circulates supercritical helium at 4.7K for cooling the CSM and 50K helium gas for cooling the current leads. The electrical system includes a DC power supply providing 50kA of current and a fast discharge system used for quench protection. The fast discharge system uses redundant DC breakers based on vacuum circuit breaker switching and a 1 Giga-Joule discharge resistor to absorb the stored energy in the coil with a 6s decay time. *Work was supported by UT-Battelle/ORNL under sponsorship of the US DOE under Awards 4000103039 and DE-AC05-00OR22725.
The design of the Residual Ion Dump Power Supply for
ITER Neutral Beam Injector

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The Residual Ion Dump Power Supply (RIDPS) is part of the Ground Related Power Supplies, to be manufactured by OCEM Energy Technology s.r.l. (OCEM) for the MITICA experiment and for the two ITER Heating Neutral Beam Injectors (HNBI). MITICA is the full-scale prototype of the HNBI, under construction in the PRIMA Neutral Beam Test Facility in Padua, Italy. The RIDPS is devoted to feed the plates of the electrostatic Residual Ion Dump (RID), which deflects and collects the beam residual ions after the neutralization process. The maximum average voltage of the RIDPS is 25 kV, to which can be superimposed a sinusoidal or trapezoidal alternate voltage at 50 Hz, 5 kV maximum. The voltage can be regulated from 20% to 100%, keeping a maximum ripple of ±500 V. The nominal current is 60 A, with a maximum pulse length of 1 hr. The reference design of RIDPS is based on Pulse-Step-Modulator technology, with multi-winding dry transformer and a set of power modules connected in series at the output, each composed by ac/dc conversion system, dc-link and dc/dc regulator. This solution offers several advantages: high accuracy, redundancy at module level, high dynamics and no need of large output capacitor for ripple mitigation. The latter is particularly important in this application, to limit the energy in case of arches between RID plates and subsequent damages. OCEM, who supplies the RIDPS via a procurement contract with F4E, endorsed this design approach, and designed a system with 42 water-cooled power modules, easily replaceable, each with diode bridge, LRC filter with pre-charge network, two IGBTs within single package and full digital control board. In this paper, the detailed design of both power and control section of the RIDPS is described. The expected performance is shown through simulations, reproducing both normal operation and fault conditions.
The Neutral Beam Injector (NBI) is required to inject in ITER plasma Deuteron particles which, once generated in the Ion Source (IS) polarized at -1MV, are accelerated at ground potential and then neutralized. This voltage level is very demanding for the power supply system, requiring several non-standard components. This paper describes the design status of two main NBI components: High Voltage Deck1 (HVD1) and the HVD1-TL Bushing. The former is a \(-1\,\text{MVdc}\) air-insulated Faraday cage hosting the Ion Source and Extractor Power Supplies (ISEPS) and the associated diagnostics; the latter is a \(-1\,\text{MVdc}\) feedthrough, interfaced with the SF6 insulated Transmission Line (TL) connecting the Acceleration Grid Power Supply system (AGPS) with the IS, carrying inside the ISEPS power conductors and diagnostics dedicated to the NBI Ion Source services. Both components are hosted inside a High Voltage Hall (HVH) with controlled environmental condition. The procurement started beginning 2015 and the manufacturing design will be finalized in the next months, aiming at testing the equipment in factory by end 2016. This paper presents the main design choices and the solutions adopted to comply with the challenging technical requirements. In particular, electrical field analysis has been carried out to verify the electrostatic design of components in final installation conditions inside the HVH, highlighting possible critical parts in surrounding elements (such as HVH irregularities and/or protrusions) that may require application of electrostatic screens, while mechanical analysis investigates the capability of the HVD1 structure and the HVD1-TL Bushing to withstand the seismic spectra foreseen at the installation sites. Moreover the thermal aspects concerning cooling down of the ISEPS components inside the HVD1 and of power conductors inside the HVD1-TL Bushing are described. Finally, next steps in terms of definition of special tests to verify some design choices will be also reported and discussed.
Modeling of ITER TF cooling system through 2D thermal analyses and enthalpy balance

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The winding pack of the ITER Toroidal Field (TF) coils is composed of 134 turns of Nb3Sn Cable in Conduit Conductor (CICCs) wound in 7 double pancakes and cooled by supercritical helium (He) at cryogenic temperature. The cooling of the Stainless Steel (SS) case supporting the winding pack is guaranteed by He circulation in 74 parallel channels. A 2D approach to compute the temperature distribution in the ITER TF winding pack is here proposed. The TF is divided in 32 poloidal segments, for each segment the corresponding 2D model is built and a thermal analysis is performed applying the corresponding nuclear heating computed with MNCP code considering the latest design updates, such as thickness increase of the blanket shield module. The Heat Transfer coefficient (HTC) of the He flowing in the CICC and in the cooling channels of the SS case is computed with Dittus Boelter correlation at the nominal inlet pressure of 6bar. The He is assumed to enter the coil at 4.5 K in the lower terminal junction, then the bulk temperature in all the CICCs in each of the 32 segments is calculated by means of enthalpy balance between segments, considering the actual direction of He circulation, i.e. clockwise or counter-clockwise in neighboring pancakes. The He properties needed to compute the HTC, such as viscosity, specific heat and thermal conductivity, are also varied using the same strategy. With these assumptions, He temperatures close to 5.7K are computed, due to the high values of nuclear heating (which is estimated as high as 21.58 kW in the 18 TF). In the paper, the methodology is presented and the results are discussed in detail. Further parametric analyses are also presented to show the impact of the inlet temperature and of the nuclear heating on the temperature distribution.
Energy dissipating resistors for the ITER switching network units

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The superconductive coils of ITER magnet system will be energized by ac/dc converters. Before each plasma pulse the magnet system will be pre-charged with energy (8GJ) to be used for generating the toroidal loop voltage required for the gas mixture breakdown and plasma formation. This will be realized by inserting energy dissipating resistors in series with the central solenoid (CS) modules and two poloidal field (PF) coils, PF1 and PF6, with the help of circuit breakers of switching network units (SNU). The dissipating resistors will be located in a stand-alone building with the required cooling conditions and connected with the circuit breakers by cables. The compact design of the switching network resistors (SNR) is provided by modular approach, effective system of forced ventilation and the ability to withstand high-thermal loads at the temperatures up to 350 °C. The resistive element is made from stainless steel plates in a zigzag pattern to minimize self-inductance, thus decreasing the switching overvoltages during transients. Selection of material with a low-temperature resistance coefficient restricts SNR resistance variation, which is necessary to ensure the required plasma ignition conditions. The main ratings of the resistors correspond to the following operational parameters: maximum current and voltage at breakdown are 45 kA and 8.5 kV, respectively, dissipated energy per coil system in normal operation is up to 1 GJ. This paper describes the resistor design based on the relevant analysis and the detailed investigation carried out during R&D phase, as well as the procedure and results of type tests on a full-scale prototype. In particular, considerable attention is paid to thermal and pulsed current tests with the aim to prevent steel plates deformation caused by heating and high currents. The successful results of the tests confirmed the suitability of the resistor design and compliance with ITER requirements.
Type Tests of Switches for the ITER Coil Power Supply System

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High current DC switches play a very important role in the ITER coil power supply system (CPSS) being key components of its two major parts: switching network units (SNU) for plasma initiation and fast discharge units (FDU) for superconducting coils energy extraction in case of quench. For both functions, circuit-breakers rated up to 70 kA steady-state current and 10 kV voltage are required to transfer the coil current into discharge resistors. Besides, make switches with similar ratings will be used in SNUs to reduce the resistance and, hence, the loop voltage during plasma initiation. Moreover, make switches are required for protection of CPSS components. Two groups of switches were developed in Efremov Institute for the ITER project. The first group involves three types of devices with considerable lifetime: open, make and disconnect switches with similar design and ratings, which are intended for SNU. The second group comprises two very reliable explosively actuated switches (circuit breaker and make switch), that will be used as back-up protective devices in FDU. Being mechanical devices, all these switches are characterised by extremely fast operation: 2-4 ms for SNU switches and less than 1 ms for FDU back-up switches. After a short description of switch design, the paper focuses on the procedure and results of type tests on full-scale prototypes, which were manufactured after completion of the preliminary design at the beginning of 2012. The test program that was implemented in 2013-2015, among a number of electrical, hydraulic and functional tests, included life-time tests with rated currents, mechanical endurance tests for SNU switches and reliability tests for backup circuit-breaker. The successful results of the type tests confirmed the suitability of switch design and compliance with ITER requirements and made it possible to start manufacturing of the switches for delivery to the ITER Site.
On optimization of air cooling system of FDR dissipating energy from ITER magnet coils

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The Fast Discharge Resistors (FDR) under development at NIIEFA are intended together with switching equipment to dissipate energy released in case of quench of the ITER superconducting coils, thereby protecting them against failure. FDRs are made of sections consisting of 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\textdegree C practically adiabatically. The resistors should be cooled to their initial temperature within 4-6 hours. For this purpose the air cooling system based on the use of natural air circulation in the system of channels formed by supply and return pipes, vertical modules and chimneys has been developed. 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. Contrary to the expectations the proposed measures on optimizing the air cooling system, while mitigating the negative effect of air flow non-uniformity in the FDR modules, did not provide the specified 4-6 hours for cooling of the resistors. Therefore, despite evident advantages of the natural air circulation used for cooling of the resistors (saving of equipment, space and energy) the authors were compelled to consider the resistor cooling system based on the forced air circulation produced by blowers. The reported analysis continues previously performed studies of the FDR cooling system. The idea to apply the forced air circulation in the resistor cooling system and to install diaphragms in each module to equalize air flow in the cooled-in-parallel modules made it possible to reduce the time for their cooling to the specified values without a considerable reconfiguration of the air cooling system.
NSTX-U coils bus bars design and construction

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NSTX-U COILS BUS BARS DESIGN AND CONSTRUCTION

The construction of the NSTX upgrade project was completed in the fall of 2015. The multi-year capital project was budgeted at $94 Million. The reactor will be used to run experiments under increased Toroidal Field (TF), Plasma Current (Ip), Beam Injection Power, and pulse length. The Bus Bars connect the magnetic coils to the power supply lines. The bus bars design consists of co-axial, water-cooled and air-cooled bus bar systems. The bus bars design was analyzed and satisfied the NSTX structural design criteria. FEM analysis was performed using ANSYS software to verify the performance of the bus bars under the increased current loads. The processes used for fabricating the bus bars include forming, machining, brazing, welding and water-jet cutting. Individual conductors were insulated using Kapton Tapes for electrical insulation and Fiber Glass Wetted with Epoxy Hysol to provide further electrical insulation and a protective mechanical coating. The joint surfaces were silver plated and the bolts torqued appropriately to maintain joint resistances within an acceptable range. Structural supports were provided as necessary to counter forces against the bus bars due to the magnetic fields, short circuit conditions and thermal boundary conditions. The insulated bus bars and assemblies were hi-pot tested to verify insulation; the joints were resistance checked; and the water-cooled buses were leak checked using hydrostatic pressure testing. At completion of the bus bars installation, pre-operational testing was performed to verify that the coils bus systems are capable of meeting the required current and voltage ratings. *This work supported by the US DOE Contract No. DE-AC02-09CH11466
Tungsten and Iron sputtering properties over recrystallization temperature

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Sputtering properties of tungsten (W) should be evaluated correctly for lifetime estimation of divertor components. Especially, at elevated temperatures, recrystallization would cause grain structure reconstruction, which would influence sputtering properties and surface morphology changes. However, the detailed studies haven’t been performed. Actually, the temperature of divertor could increase by slow transients to ~2,000 °C or more, which is higher than recrystallization temperature of W (~1,200 °C). In the previous study, the effect of recrystallization on sputtering was studied on iron (Fe). It was observed that grain orientation of Fe changes into that of small sputtering yield [1]. However, this effect hasn’t been investigated on W. In this study, the sputtering properties of W over recrystallization temperature is studied in detail. In addition, prior to the study of W, we have also investigated the sputtering properties and crystalline structure changes of Fe since Fe has the same bcc structure as W, which could give us important information on high temperature sputtering. We performed deuterium (D) irradiation experiment to Fe. The irradiation energy and flux were 1 keV and ~10^{20} D/m²/s. The sample temperature was varied between 473-973 K. The sputtering yield was estimated by mass loss measurements. After the irradiation, surface morphology were observed using a scanning electron microscope (SEM). The sputtering yield of Fe by D irradiation is increased over ~720 K, which almost corresponds to the recrystallization temperature of Fe. The sputtering yield of 973K is about 1.4 times of 473 K. From observation of the surface morphology, surface roughness grew with increasing of temperature along with the sputtering yield and reached a few hundred nanometer in 973 K. We’ll also investigate the sputtering properties of W and compare with the results of Fe. [1] Y. Ueda et al., J. Nucl. Mater. 386-391 (2007) 367-370
Measurement of tungsten optical absorption rate and dynamics of W melting behavior

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Currently, in regard to the plasma facing material, Tungsten (W) is a major candidate at ITER. A recent study has been reported indicating that the transient thermal load such as ELM or disruption causes metal surface melting or evaporation of W. However, the property and behavior of the W above the melting point has not yet been sufficiently known, and many of the previous studies are postmortem analyses. Thus in-situ observation of melting W is important. In this study, a pyrometer measurement and a construction of two-dimensional (2D) temperature distribution measurement for the laser-melted W were performed to clarify the behavior of the W above the melting point. In the experiment, a Nd:YAG laser (1,064 nm, irradiation diameter 0.6 mm, maximum power 7 kW) has been used as a heat source that simulates a non-steady-state heat load, and the mirror-polished (surface roughness Ra 0.01\textmu m) W samples (by A.L.T.M. Corp.) were irradiated. By measuring the temperature rise of the entire sample using a welded thermocouple, the energy absorption rate was measured. In order to obtain the energy absorption rate at a temperature above the melting point, time-varying laser irradiation pulses were applied. The surface temperature dependency of the energy absorption rate was then determined. To investigate the W melt behavior in more detail, a 2D temperature measurement is being constructed. In the system, the black body radiation from the surface was delivered through half mirrors and band pass filters, to the sensor. The 2D temperature profile can be obtained by analysis of the two-color images.
Molten layer characteristics of W materials and its protective coatings by pulsed laser irradiation

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Tungsten (W) is a primary candidate of plasma-facing materials for fusion reactors. But erosion due to melting and evaporation of W caused by transient heat loads are concerned. A pulsed laser simulating the transient heat loads was irradiated to three tungsten materials and the behavior of the molten layer was investigated. In addition, aluminum (Al) and tin (Sn) was deposited on W and the effects of the protective film was investigated. Plate samples were installed in an ultrahigh vacuum and irradiated by Nd-YAG laser (wavelength 1.064 nm) onto the circular spot of a diameter of 0.6 mm. We prepared three W materials; pure W, W-10%Re and W-2%Ta. We observed samples after the heat load irradiation with a laser microscope for surface profiling. In addition, thin films (1-3 μm) of Al and Sn were deposited on W samples using a magnetron sputtering device as protective films. Evolution of surface temperature was measured by a using two-color thermometer. After irradiation in 2.33-3.68 GW/m² to a pure W sample, the central part became dented, however, after irradiation in higher heat fluxes of 4.14-5.15 GW/m², the central part protruded but not droplet ejection was observed. In contrast, the central part of irradiated W-2%Ta became dented deeply in wider range with increase in the power density, which could be attributed to the release of Ta droplets by bumping, closely related to microstructure and/or impurity concentration determined by the materials production processes. By using this laser melting technique, it is possible to evaluate molten layers behavior of W materials by giving laser heat loads and observing melting spots in detail.
Secondary radiation damage and gas production in plasma facing materials under fusion neutron irradiation

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Primary radiation damage (atomic displacements) and Helium and Hydrogen production rates in plasma facing components (PFCs) of a fusion system are usually determined by the high energy parts of neutron spectra formed in plasma chamber from the initial fusion neutron source. According to presented estimates, the energetic alphas and protons, appearing in PFC materials in the \((n,a)\) and \((n,p)\) threshold reactions, may cause additional, secondary material damage increasing damage from neutrons by a factor of 2-4. An additional Helium production is expected in systems with a soft neutron spectrum in case of using W- or an W-alloy as an armor attached to the first wall. Calculations show that Re and Os as transmutation products appeared in W under long term neutron irradiation account for a source of intensive helium production in \((n,a)\) reactions with these secondary radionuclides. These even if preliminary estimated radiation effects should be taken into account in new material considerations for long term concepts.
Microstructural and micro-mechanical changes in tungsten under high flux plasma exposure

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Recent theoretical and subsequent experimental studies suggest that the uptake and release of deuterium (D) in tungsten (W) under high flux plasma exposure (i.e. under ITER-relevant conditions) is controlled by dislocation microstructure induced by the plasma itself. A comprehensive mechanism for the nucleation and growth of D bubbles on dislocation network under high flux low-energy plasma exposure was proposed and validated. The process of bubble nucleation can be described as D atom trapping at a dislocation line, its in-core migration, the coalescence of several D atoms into a multiple cluster, which eventually transforms into a nano-bubble by punching out matrix atoms on the dislocation line. This view implies that the initial microstructure might be crucial for D uptake and degradation of the sub-surface layer under prolonged plasma exposure. Understanding of the role played by the initial microstructure is the purpose of this work. In this work, we apply several experimental techniques to investigate the microstructure and mechanical properties of surface and sub-surface layer of W exposed to the high flux plasma. In particular, we use transmission and scanning electron microscopy, as well as nano-indentation measurements. To reveal the impact of the initial microstructure, we have performed exposures in single crystal, poly-crystal and heavily deformed polycrystal tungsten samples. The preliminary TEM study demonstrates that even in single crystal sample, high flux plasma exposure induces high density of dislocations and tangles in the sub-surface area. The presence of the plasma-induced microstructure is well detected by the nano-indentation experiments, which provide reach information about change of material hardness and depth distribution of the irradiation-induced microstructure.
We investigate the ablation characteristics of plasma facing materials (PFM) using thermal plasma facilities. A high enthalpy, 400 kW plasma testing facility which uses an enhanced segmented arc torch as a plasma source and 55 kW vacuum plasma spraying system produce particle flux greater than $10^{24}/(m^2/sec)$ and heat flux greater than 10 MW/m$^2$, levels that are relevant for testing the PFM under fusion reactor conditions. We measure the ablation rate and surface roughness change that result from the ablation and perform a morphological analysis of the PFMs before and after the ablation test to investigate the ablation characteristics. We also perform heat flux test with an electron beam facility which impose the heat flux and compare the result with the ablation experiments to identify an effect of the particle loads on the PFMs.
Helium ion irradiation of tungsten carbide neutron shields

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High-field spherical tokamaks may be a viable technology for relatively compact fusion power devices (Costley et al. Nucl. Fus. 2015). However, such reactors leave little space for shielding of the central column, which must protect the inner superconducting magnets from high energy neutrons. Tungsten carbide cermets are promising candidate materials for such shields: They have high thermal conductivity, satisfactory oxidation properties, and can be manufactured at relatively low temperatures in complicated geometries. Furthermore, the neutronics properties of such cermets are very encouraging (Windsor et al. Nucl. Fus. 2015). However a particular concern is the production of helium ions via (n,α) transmutation reactions under high-energy neutron irradiation, which could lead to stabilisation of void embryos and bubble formation resulting in potential structural degradation. In this study we simulate the production of helium gas by implanting helium ions into cermet thin foils. The foils are made from tungsten carbide cermet with an iron-chromium metallic binder. Their microstructural evolution under 6 keV helium irradiation is tracked in-situ through transmission electron microscopy. A range of fluences and temperatures were investigated: from 0 to the order 10^{17} ions/cm^2, and from room temperature to 750 °C. Implantation led to the formation of nanoscale bubbles in the foils, both in the major carbide phase and in the minor metallic binder phase. In general, bubbles in the WC phase were very small, e.g. on the order of 1-2 nm, while in the binder such defects were typically much larger. Interestingly, at carbide-metal phase boundaries we observe bubble coalescence, which is particularly prominent at low temperatures and very high fluences. Such bubble coalescence has not yet been reported and may adversely affect bulk mechanical properties. Our systematic work in quantifying these effects as a function of irradiation conditions is therefore particularly needed.
Production of tungsten particles from a radiofrequency plasma jet

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Controlled fusion research represents an important step for sustainable energy production once with the development of the International Thermonuclear Experimental Reactor (ITER). ITER proposes a deuterium - tritium fusion reaction for hot plasma creation. During plasma-wall interactions, small tungsten particles, from nm to microns will be produced in the tokamak chamber. These particles can incorporate tritium leading to potential danger of environment contamination and also explosions. Thus, it is of interest to produce, aiming to study their properties in advance, tungsten particles with plasma methods. In this line, our contribution consists in analysing the production of tungsten particles, by using a radiofrequency plasma in hollow electrode discharge configuration. As a main gas we have used argon, the cathode was from tungsten and the plasma was focused in a small tungsten pipe, with the inner diameter from about 2 mm. Small tungsten particles were produced, as example in using 1500 sccm gas flow rate and for 225 watts, input power. The shape, size and size distributions of the resulted particles were investigated by Scanning Electron Microscopy, while EDS measurements demonstrated that the particles were from tungsten. The tungsten particles dimension and number can be modified by varying the process parameters, like gas flow rate, input power and pressure. Acknowledgements V. Marascu acknowledges the support in the frame of the project 1-EU12 WPEDU-RO, “EUROfusion Consortium contribution to education in fusion research at the predoctoral and PhD level “.
Thermal management of tungsten leading edges in DIII-D and ITER

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Power exhaust is perhaps foremost among the issues for ITER and post-ITER devices, as well as for existing large confinement devices as they increase power. A related concern is the alignment of plasma facing components to avoid protruding (leading) edges that would intercept field lines and incur very high loads and high erosion. This concern prompted the transient melt experiment in JET, followed by additional ITER-coordinated W (tungsten) leading edge experiments first in DIII-D and then in EAST. Alignment is a concern also in the DIII-D Metal Tile Experiment planned in the 2016 campaign. The focus, on high-Z impurity transport in the plasma edge, is part of a broader long range goal to develop and test advanced divertor configurations and validate reactor-relevant materials. Each tile in two toroidal rings of divertor tiles, one on the shelf, another ring on the floor, will have a W-coated TZM (molybdenum alloy) insert spanning its toroidal width. The TZM inserts are 50 mm wide radially, 9.5 mm thick, bolted from the back into wide slots in newly made carbon tiles. Adequate thermal management includes well aligned tiles (specifications plus experience with installation); filleted leading edges; and thermal modeling showing acceptable temperatures and stresses for representative heat loads for L and H mode plasmas and with ELMs. The team considered several approaches (fish-scale, roof top, chamfer, fillet) to minimize power to any leading edges. Thermal modeling of a flat insert/tile with a 2-mm fillet gave $T_{\text{max}}$ on the insert surface of 745°C at the end of a 3.5 s shot for an insert 0.3 mm above its neighbor, strike point with 6.5 MW/m$^2$ peak toward the inside of the insert, and 2.5° angle of incidence. ELMs (10X power for 2 ms) brought a very small area on the leading edge to ~1500°C. This work is supported by the U.S. DOE under DE-AC04-94AL85000\textsuperscript{1,1} and DE-FC02-04ER54917\textsuperscript{4,4}, DE-AC05-00OR22725\textsuperscript{5,5}
Study of deuterium retention in Be-W coatings with distinct morphologies

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Migration of impurities during ITER plasma discharges will result in the formation of co-deposited mixed materials on the surface of plasma facing components (PFC) with properties distinct from those of the original PFC. These issues have motivated the fusion community to investigate Be-W coatings, in particular their fuel retention behaviour, since in ITER the deposits will present a beryllium (Be) and tungsten (W) matrix. Also recent investigations have shown that the erosion and deposition rates of plasma impurities in the exposed surfaces highly depend on their morphologies [1]. On the other hand the chemical effect can hinder the individual role of superficial roughness in the fuel retention mechanisms and must be investigated in dedicated irradiation experiments involving specific Be-W stoichiometries. It is assumed that fuel retention is lower in pure W layers than in Be. Nevertheless, previous irradiation experiments carried out in a wide range of compositions evidenced already lower deuterium retention rates in Be:W (50:50) coatings presenting a smooth topography. The goal of the present investigation is to correlate the retention rate of deuterium implanted in different sets of Be-W coatings presenting specific compositions and distinct roughness morphologies imposed by the thermionic vacuum arc method procedure [2,3]. Energetic $^2\text{H}^+$ ions will be implanted up to fluences of $5\times10^{17}$ ions/cm$^2$ in selected samples. Before and after implantation, the topography and chemical composition of the samples will be studied by atomic force and electron microscopies and by ion beam technics. [1] H. Bergsåker et al., J Nucl. Mater. 463 (2015) 956. [2] A. Anghel, I. Mustata, C. Porosnicu, C.P. Lungu, J. Nucl. Mater. 385 (2009) 242. [3] A. Anghel et al., Nucl. Instr. Meth. Phys. Res. B 267 (2009) 426. Topic of the abstract in the SOFT 2016 conference: F - Plasma Facing Components
Tungsten covered carbon materials due to good thermal conductivity of carbon based materials (up to ~250 Wm\(^{-1}\)-1K\(^{-1}\)-1 for carbon fiber composites [1]) are suitable for use in fusion devices, like ITER (International Thermonuclear Experimental Reactor) [2], as divertor materials. However, during the plasma wall interactions, erosion and re-deposition, as well as formation of redeposited coatings on the plasma facing surfaces leads to retention and accumulation of fusion fuel, including tritium in the deposited layers and bulk of materials, causes shortening the lifetime of materials and modifications of the their properties. In order to estimate forms of compounds created during plasma-wall interactions in fusion reactors, it is important to understand processes taking place during high energy influence to divertor materials. Understanding of the induced modification mechanisms taking place during irradiation will allow to better estimate ways of tritium accumulation and formation of dust and flakes in fusion reactor. In order to simulate effects of the high energy (up to several GW/cm\(^{-2}\)-2) fluxes possible in the tokamak reactors, prototype deposited coatings created by Thermionic Vacuum Arc technique (TVA) were irradiated with single and multiple fs laser pulses. Changes in beryllium crystalline structure as well as in graphite crystallite sizes and amount of sp\(^3\)-3 hybridized carbon were observed during single and multiple laser pulses. Laser irradiation induces surface structural and morphological changes by ripples formations. Laser and plasma affected zones were characterized by Raman and Fourier transformation infrared (FT-IR) and X-ray fluorescence spectroscopy, as well as with SEM and EDX analysis and some interpretations of the induced modification mechanisms are presented.

Acknowledgments

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Material qualification of tungsten fibre-reinforced tungsten composite by means of tension tests

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Tungsten is a promising plasma facing material for future fusion reactors due to its unique property combination such as low sputter yield, high melting point and low activation. The main drawbacks for the use of pure tungsten are the brittleness below the ductile-to-brittle transition temperature and the embrittlement during operation e.g. by overheating and neutron irradiation. This limitation is mitigated by using tungsten fibre-reinforced tungsten composite (WfW) which utilizes extrinsic mechanisms to improve the toughness similar to ceramic fibre-reinforced ceramics. It was shown that this idea in principle works in the as-fabricated WfW as well as in the embrittled material. Recently a novel chemical vapour deposition process was developed allowing the production of large and reproducible samples [Riesch2016]. In this contribution we present a qualification program based on tensile tests on improved material samples produced with this new process. The material parameters were evaluated by means of displacement controlled “standard” tension test as well as low cycle fatigue (LCF) on as-fabricated and on embrittled samples. Standard tension tests give insight on the ultimate tensile strength and reveal the active toughening mechanisms provided by the fibres within the composite. However the expected loads on the material in a future fusion device are not only high steady state temperature but also extreme energy transients resulting in thermal cyclic loading of the material. Similar loading conditions are achieved in a low cycle fatigue test. In the as-fabricated condition samples the material is still able to bear rising load despite multiple matrix cracks. Fibre necking as well as fibre pull out was observed leading to the typical pseudo ductile behavior of the composite. The description of the mechanical tests will be supplemented by detailed microstructural investigations. [Riesch2016] Riesch et al., Proceedings of the 17th ICFRM, Nuclear Materials & Energy, 2016, submitted
The emissivity of tungsten coatings deposited on carbon materials for fusion applications

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Tungsten coatings deposited on carbon materials such as carbon fibre composite (CFC) or fine grain graphite (FGG) are currently used in fusion devices as armour for plasma facing components (PFC). About 1800 CFC tiles were W-coated for the ITER-like Wall at JET and more than 1300 FGG tiles were coated for the ASDEX Upgrade tokamak. At present the W coating production is on going for the first lower divertor of WEST. The emissivity of W coatings is a key parameter required by protection systems of the W-coated PFC and also by many diagnostic tools in order to get correct values of temperature, heat loading, etc. The emissivity of tungsten is rather well known, but the literature data refer to bulk tungsten or tungsten foils and not to coatings deposited on carbon materials. W coatings of 10 μm or 20 μm were deposited on tubes (Φ16x85x0.8 mm) made of FGG, and CFC. A hole with a diameter of 2 mm was drilled in the middle of the tube perpendicular to its axis. This hole played the role of a black body. The W-coated tube was heated up to 1200 °C by electric conduction. The emissivity was measured at the wavelengths of 1.064 μm, 1.75 μm, 3.75 μm, 4.0 μm and 4.25 μm using IR detectors. It was found that the structure of the substrate, particularly in the case of porous CFC, has a significant influence on the emissivity values. The temperature dependence of the emissivity in the range of 600°C-1200°C and the influence of the viewing angle were investigated as well. The emissivity depends significantly on the investigating wavelength. For example 10 μm W coated Dunlop CFC at 1000°C has the emissivity of 0.63±0.07 at the wavelength of 1.064 μm while at 4.0 μm the emissivity drops to 0.25±0.07.
Dependence of damage depth profile on deuterium retention in C+ implanted W

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Tungsten (W) is a candidate for plasma facing materials in D-T fusion reactors due to its higher melting point and lower sputtering yield. During the plasma operation, W will be exposed to energetic particles including hydrogen isotopes, neutrons, and impurities like carbon (C). It is well known that hydrogen isotopes are trapped in the defects produced by the energetic particle irradiation. In addition, a W-C mixed layer suppresses the deuterium (D) diffusion. Therefore, it is important to evaluate the synergetic Fe2+2+ and C+ implantation effect on D retention behavior for W under various damage distribution profiles. In this study, the irradiation damages were introduced by 6 MeV Fe2+2+ implantation with the damage concentration of 0.01, 0.1, and 1 dpa (displacement per atom). Then, 10 keV C+ implantation for these samples was performed with the damage concentration of 1.14 and 11.4 dpa. According to SRIM, implantation depth of Fe2+2+ and C+ is about 1.5 µm and 0.05 µm, respectively. Thereafter, 3 keV D2+ was implanted with the ion fluence of 1.0 × 1022 m−2, and thermal desorption spectroscopy (TDS) measurements were performed to evaluate the D2 desorption behavior. The experimental results indicated that C+ implantation enhanced D retention trapped by vacancies, but D retention was diminished in C+ implanted W with higher fluence due to a decrease of the number of D trapping site as C+ trapped by vacancies increased. Additionally, it was also found that D was trapped by the high density of dislocation loops introduced by C+ implantation, and D retention behavior adsorbed from the surface was not controlled by Fe2+2+ implantation damage level. However, D was diffused toward the bulk and was trapped by stable voids introduced by Fe2+2+ implantation even if the dense damages were introduced by C+ implantation near the surface.
Profile analysis of the first wall heat load required for demo blanket concept

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Understanding of the heat load profile on the first wall is one of the key issues to establish the DEMO blanket concept, because the thermal stress on the each blanket module depends on its surface heat load, and it will vary with the first wall shape, the toroidal/poloidal position and the plasma equilibrium. Thus, the first wall surface of the blanket module has to be designed according to the heat load profile on the first wall. One of the main factors of the heat load is plasma heat flux. Some of plasma particles in the SOL (Scrape Off Layer) region go to the first wall along the magnetic field line. It is known that this parallel heat flux by the plasma particle at the outboard mid-plane decays exponentially in the radial direction and its decay length depends on the connection length. In the divertor phase, the magnetic field line is divided by the first wall, i.e. the connection length and the decay length change. This effect makes it difficult to estimate the heat load. Here, the new calculation model has been developed. From the 2 dimensional magnetic field line trace data, the SOL region is divided into the several flux tube regions, and the decay length and the parallel heat flux in each region is calculated, and the heat load is estimated. In the case of R = 8.5m and fusion power is 1.4GW Japanese DEMO reactor, the heat load is peaked near the inboard baffle plate and its value is 0.9MW/m². In this presentation, we'll suggest the new first wall shape design concept which decreases the peak value of the first wall heat load from the more detailed analysis with the 3 dimensional magnetic field line trace data.
Thermal performance augmentation by detached rib-arrays for helium-gas cooled First Wall applications

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Rib-roughening the helium-gas cooled channels in plasma-facing components of DEMO (First Wall (FW), limiters or the divertor) enhances heat transfer and reduces structural material operation temperatures. The rib-elements induce a three-dimensional, unsteady flow field and heat transfer is augmented by mixing the fluid in the near wall regions and boundary layers. Whereas the overall heat transfer increases at rib-roughened channel wall, local decrease in heat transfer occurs at flow stagnation regions in the vicinity of rib-channel-wall junction. The present study examines the applicability of rib-arrays detached from the plasma-facing cooling channel wall for homogenizing the surface temperatures and raising the thermal efficiency within the helium-gas FW cooling concept. Furthermore, detached rib-arrays are expected to be advantageous in terms of cost-effective fabrication and local stresses compared to wall mounted ribs. Heat transfer and flow characteristics were computed by Detached-Eddy-Simulations (DES) at Reynolds numbers of ReDh=5E4 to ReDh=1.5E5 (corresponding to helium mass flow rates of 0.026 kg/s to 0.072 kg/s at 8MPa pressure). Thermal-hydraulics for rib-arrays of square rib-elements with a cross section of 1 mm x 1 mm and a clearance to the channel wall of c =0.1 mm, 0.3 mm and 0.5 mm were investigated. The channel cross section was 15 mm x 15 mm with round-edges of 2 mm radius, the rib-pitch-to-rib-height-ratio was p/e=10 and the rib-height-to-hydraulic-diameter-ratio was e/Dh=0.0653. A constant heat flux density of 750 kW/m² ±250 kW/m² and 80kW/m²² respectively were applied at the plasma-facing and breeder-unit-facing FW structural surface. The results show, that for increasing clearance-to-rib-height-ratios (a) peak values are reduced leading to homogenized surface temperatures and (b) the thermal performance factor for increased heat transfer decreases. For all simulations, mass flow rate dependent correlations for heat transfer coefficient and pressure drop prediction were derived.
Thermo-mechanical analyses and ways of optimization of the helium-cooled DEMO First-Wall under RCC-MRx rules

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The EUROfusion Consortium develops a design of a fusion power demonstrator plant (DEMO) in the framework of the European “Horizon 2020” innovation and research program. One of the key components in the fusion reactor is the breeding blanket surrounding the plasma, ensuring tritium self-sufficiency, heat removal for conversion into electricity, and neutron shielding. Among the 4 candidates for the DEMO Breeding Blanket, 2 of them use helium as coolant, and another one uses helium to cool down the FW only. Due to uncertainties regarding the plasma Heat Flux load the DEMO Breeding Blanket integrated First Wall will have to cope with, a set of sensitive thermal and thermo-mechanical analyses have been performed in order to define the possible margin against HF the integrated Helium Cooled Eurofer FW could have. Based on the Helium Cooled Lithium Lead (HCLL) equatorial outboard module dimensions, thermal and stress FEM analyses have been performed with Cast3M with various FW front wall thicknesses and Heat Flux, under normal steady state condition. Stress have been analysed with RCC-MRx code including high temperature (creep), cyclic (fatigue) and irradiated rules. This paper shows that the thickness of the plasma-facing wall of the FW should be minimized, within the limits necessary to withstand primary stresses, in order to reduce the temperature on the structure and thus prevent fatigue and creep damage as well as a reduction of the stress limit Sm which is function of temperature to prevent ratcheting. Moreover, the paper will discuss the importance of having constant HF during the reactor operation. A small variation of HF could increase a lot the risk of damage such as fatigue and creep. At the end, the effect of irradiation shows up to be the limiting criterion and penalizes the capacity of the FW to withstand high HF.
Characterization of the cooling channels of First Wall Mock-up dedicated to the HCPB-TBM qualification

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The First Wall (FW) of the EU Helium Cooled Pebble Bed (HCPB) Test Blanket Module (TBM) faces the fusion plasma and experiences high heat fluxes; therefore its cooling channels design is a key R&D task for qualifying the HCPB TBM for the fusion reactors ITER and DEMO. Within the manufacturing and qualification activities performed in KIT for the HCPB TBM, a First Wall Mock-up (FWM) was designed and manufactured. The objective of this study is to characterize the hydrodynamic behaviour of the FWM cooling channels by investigating the coolant pressure loss for each channel. The FWM has a shape of rectangular prism (710 mm × 405 mm × 45 mm) and 10 U-shaped cooling channels which have a square cross section (15 mm × 15 mm) with rounded corners of 4 mm radius. The FWM was integrated into an experimental gas loop that has the relevant instrumentation and piping system for measuring and controlling the gas (e.g. compressed air of 6 bar abs.) flow parameters. This paper presents the experimental results of measuring the pressure loss across the FWM cooling channels and discusses its main causes. In addition, the experimental results are compared with theoretical values obtained from relevant models and formula available in the literature. The results of this study will support the qualification of the HCPB TBM mock-ups which will be tested at fusion-relevant heat flux and helium cooling in the Helium Loop Karlsruhe (HELOKA) facility.
Optimization of first wall helium cooling system of European DCLL using CFD approach

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Dual Coolant Lithium Lead (DCLL) is one of the four breeding blanket concepts being developed within the EUROfusion project as candidates for the European DEMO. One of the most challenging components of breeding blanket in terms of thermal-hydraulic is a first wall. In order to handle the high thermal loads that the DCLL first wall is facing a proper design of a helium cooling system is crucial. The present work deals with evaluation of the first wall cooling ability under the DEMO conditions and optimization of geometric and operational parameters of the cooling system composed of helium channels. For this purpose, a sensitivity study to evaluate dependence between geometric and operational parameters was performed. In particular, effects of distance between the helium cooling channels (channel pitch) and coolant inlet velocity on the maximum EUROFER temperature inside the structural material in dependence on various conditions were assessed. These effects were detected and described in the present work to fulfill the EUROFER temperature limit by applying reasonable combinations of channel pitch and coolant inlet velocity values. All studies were performed using CFD approach. Preparation process of the CFD analyses including geometric parametrization of a computational mesh which was necessary for implementation into the CFD solver due to a high number of simulations needed to be performed in order to obtain an appropriate sensitivity study is also described.
Development of force reconstruction method on EU ITER TBM based on strain measurements

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The EU ITER Test Blanket Module (TBM) sets, which consist of TBM box and shield, will be located inside the equatorial port \#16 of ITER. One of the important objectives of the TBM program, starting from the first H-H phase, is the validation of the theoretical predictions of the structural behavior of the TBM set under thermal, mechanical and electromagnetic loads. High electromagnetic forces acting on the TBM sets, especially during off-normal operation, are one of the most demanding loading conditions. In order to estimate these forces, a force reconstruction method is proposed. The reconstruction is based on measurements of strain sensors on the attachment system, which connects TBM box and shield. Due to the distributed characteristic of the electromagnetic forces, the forces are reconstructed in terms of modal forces. This approach as well as the use of modal forces to validate electromagnetic analyses is described in detail. In addition, the development of the modal models of the TBM sets required for the force reconstruction algorithms is explained. A number of test cases has been defined that cover a wide range of ITER relevant excitations of the TBM sets due to electromagnetic forces. Based on the simulation of these test cases the influence of errors in the developed models on the accuracy of the reconstructed forces is evaluated. In this context, also the influence of the number of strain sensors is discussed. The determination of the sensor arrangements is based on a genetic algorithm. The requirements for the placement and installation of the strain sensors, which serve as input parameters for the genetic algorithm, are described. Furthermore, the development of an experiment with a TBM box mock-up with a simplified attachment system to study the reconstruction of modal forces and the related validation of FEM codes is presented.
Integral experiments of neutron transport in blanket module with hydrogen ion beam

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DEMO reactor must achieve total TBR >1 with high level of accuracy and confidence in the design process. However there is no relevant neutron sources before ITER /TBM, and even in ITER, neutron field is considerably different due to the shield blankets surrounding TBM. This study proposes verification technique to experimentally simulate reactor neutron field and evaluates its expected accuracy. A small blanket module with the titanium hydrides coated first wall can generate fusion neutron by 100keV level deuterium beam with reactor relevant spectrum and geometry for benchmark of neutronics calculation. Suitable materials for moderator and reflector surrounding the module further simulate the neutron field within the module. Tritium production profile was evaluated by a neutron transport calculation by MCNP and compared with the case of fusion reactor. A small module of 25πcm\textsuperscript{2} plasma facing surface F82H RAFM, lithium titanate for breeding material, water and Be multiplier are layered as a model. Titanium coating of 5 micrometer tritiated to TiT\textsubscript{2} generates 10^8 neutrons /cm\textsuperscript{2}s uniformly by a 100 keV, 1.3 mA/cm\textsuperscript{2} deuterium beam. Module is surrounded by 40cm thick graphite that was found to be sufficient to simulate the total neutron reflection. Neutron flux attenuates half orders in 5 cm of magnitude along with the axis, and spectrum and flux profile agreed well with that of volume neutron source and full torus blankets case. Typical tritium production profile agreed within error of 10 %. Total 9 hours irradiation was sufficient for the tritium production profile measurement in the breeder with a liquid scintillator. Difference between the point source and volume source was found to become negligible with the 25π cm\textsuperscript{2} footprint. The result of this study suggests that experimental verification of the breeding performance of the blanket is possible with small scale facility.
Analysis of Lorentz’s and Maxwell’s forces on DEMO segments under normal and off-normal conditions

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Off-normal operations in Tokamak reactors result in the induction of eddy currents that, coupled with the large magnetic field, impose strong electromagnetic forces (Lorentz’s forces) to fusion reactor components. In addition the presence of ferromagnetic material induces Maxwell’s forces as interaction between the magnetized material and the external magnetic field that are thus present also during normal operation. Maxwell’s forces, due to its nature, differs from Lorentz’s one and can give a significant contribution to the related total loads as shown, for example, in the EM analysis of the ITER TBM. For this reason a particular attention is here given to these forces that were not evaluated in similar previous works. The EM analysis of a DEMO reactor configuration, elaborated by EUROFusion in 2015, is here presented. The FEM model, developed with ANSYS-EMAG, represents a 20° sector of the DEMO machine including coils, VV and inboard (IB) and outboard (OB) segments. The segments radial segmentation is defined considering the HCPB concept. Due to the complex internal structure of each segment, simplifications in the design and material properties have been considered. The impact of the made assumptions on the final results is explained in detail. The results, obtained considering a major central disruption, show that Lorentz’s forces mainly induce a poloidal moment on the segment components approximately directed towards the plasma center. On the other hand, the Maxwell’s forces are predominantly in the radial direction towards the center of the machine following the gradient of the toroidal magnetic field as predicted by Kelvin’s formula. In particular, due to their high value (around 6 MN for a whole segment), these forces have to be carefully considered during the evaluation of the structural integrity of the segments as well as for the design of the attachment system between segment and VV.
The Tore Supra tokamak is being transformed in an x-point divertor fusion device in the frame of the WEST (W-for tungsten-Environment in Steady-state Tokamak) project, launched in support to the ITER tungsten divertor strategy. The WEST project aims to test W monoblock Plasma Facing Units (PFU) under long plasma discharge (up to 1000s), with thermal loads of the same magnitude as those expected for ITER. Therefore the divertor is a key component of the WEST project, and so is its support structure, which has to handle strong mechanical loads. The WEST upper and lower divertor are made of 12 30° sectors, each one composed of 38 PFU that can be made of tungsten, CuCrZr or graphite. A generic 316L stainless steel 30° conic support plate is used to hold the 38 PFU together, regardless of their material. The PFUs are fixed on the support plate thanks to 152 Xm19 stainless steel fixing elements (4 per PFU), and in each of this fixing element an Aluminium-Nickel-Bronze alloy (Al-Ni-Br) pin is engaged in a slotted hole, in order to allow thermal expansion in the length direction of the PFU. The support plate is fixed on the divertor coil casing thanks to 10 M10 screws. Mechanical loads which act on the PFUs are transmitted to the support plate through the fixing elements. These loads are due to Vertical Displacement Event (VDE), disruptions and thermal expansion of the PFU. First the different load cases, PFU configurations and scenario are presented. Then an ANSYS plastic mechanical simulation is performed in order to validate the number of cycles of the support plate for each scenario: 30 000 cycles in steady-state and 3000 cycles in VDE. Finally reactions forces from the previous ANSYS simulation are used in order to calculate the stress in the M10 screws.
First experiments with tungsten limiter on the T-10 tokamak

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In 2015 the graphite limiter was replaced by the tungsten one on the T-10 tokamak. The limiter was made in “Efremov Institute” from the ITER-grade “POLEMA” tungsten used for ITER divertor plates manufacturing. “POLEMA” tungsten doesn’t contain any impurities and has a high thermal conductivity and heat capacity. Tungsten has a polycrystalline structure with a grain size about 30 µm. The tungsten limiter has no active cooling. During the experimental campaign the tungsten limiter was exposed to ~ 1000 working discharges, with heat load on the limiter up to 1 MW/m². No changes of tungsten surface have been observed after such heat load. In future experiments with power ECR heating we are going to achieve a load up to 5 MW/m². Destruction of tungsten in discharges with runaway electrons was investigated. The melting of the tungsten limiter on the low-field side of torus in the equatorial plane was observed. The drops of melted tungsten and redeposited tungsten layers were observed near this area. No redeposited tungsten was detected in the rest area of the T-10 vacuum chamber. Replacement of the graphite limiter by the tungsten one didn’t lead to a significant change of the T-10 plasma parameters. Both in case of C-limiter and W-limiter the main light impurities are oxygen and carbon. In case of W-limiter the level of light impurities is 3.5-4.5% for carbon and 3-4% for oxygen which is higher by 10-20% in comparison with C-limiter discharges. Zeff is of 4-5. The Te(r) profile and peripheral radiation determined by light impurities changed slightly in comparison with the C-limiter discharges. Radiation losses from the center, determined by tungsten is highly increased with peaked Pr profile. The value of the W concentration in the T-10 plasma was estimated.
Status and plans of in-situ diagnostics based on synchrotron radiation scattering at station “Plasma”

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The residual mechanical deformation and stress were measured in the preliminary experiments carried out at synchrotron radiation (SR) scattering stations on VEPP-3 in the Siberian Center of Synchrotron and Terahertz Radiation. Significant changes in the SR diffraction are found as the result of material recrystallization or irradiation of the material by plasma or high energy ions. It implies that the SR scattering diagnostics may be an informative instrument for in-situ observations of state of plasma facing components. The next step of the SR scattering diagnostics development at the novel scattering station “Plasma” is the dynamic measurements during pulsed heat loads. Currently 1J YAG laser is used for the 0.2ms heat load simulation and 100J laser is under development. The destructive effect of pulsed heat loads is caused by mechanical stresses in a strongly non-uniformly heated material. The main aim of current development of diagnostics based on SR scattering is the dynamic measurements of deformation and stress dependences on the depth under the surface. The deformation and stress distributions may be calculated using measurements of the diffraction peak of SR passed through the sample. The set of requirements determines restrictions on SR brightness and energy. The SR from VEPP-4 with energy 69keV will be used for experiments with tungsten. Also a single crystal samples are necessary for increasing of the diffraction peak brightness. Currently 1D gas X-ray detector DIMEX is used for measurements. The development of silicon detector is in progress. The diffraction peak parameters for SR reflected by germanium single crystal were observed in the first dynamic experiments. The changes of intensity and position of the diffraction peak were measured during the laser irradiation of material. The result demonstrated possibility of the dynamic experiments. Measurements of the diffraction peak passed through the tungsten sample will be the next step.
Dust remobilization experiments on the COMPASS tokamak

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Dust transport is among important issues for ITER and DEMO, where material erosion will be significant. One of possible mechanisms how material is eroded from plasma facing surfaces is the remobilization of dust particles linked to their lifetime there and to the formation of dust accumulation sites. On the COMPASS tokamak, dust remobilization experiments have been performed using a tungsten surface with well-defined seeded dust particles exposed to L-mode and ELMy H-mode discharges as well as to a disruption. On small flat tungsten blocks, a dust of the particle size up to 25 mm was deposited by a low speed gas gun. Dust particles were prepared as sub-millimetre spots positioned on the top and on the side surfaces of the block. Dust particle positions for all prepared spots were mapped using a scanning electron microscope (SEM). Then, the samples were mounted on a manipulator allowing insertion to a fixed position close to the divertor region and were exposed to a discharge of the COMPASS tokamak. After the removal of the blocks from the vacuum vessel, positions of the seeded tungsten dust particles were again mapped by SEM and compared with the original ones, deducing movements of individual particles as well as a pattern-like behaviour. Remobilization of dust particles were monitored by a fast visible light camera with a pixel resolution of 0.3 mm at 15-40 kfps. Heat flux conditions were derived from measurements of the divertor probes (0.02-0.5 MW/m²2 in L-mode and inter-ELM periods of H-mode) and Langmuir and ball-pen probes (<10 MW/m²2 during ELMs). The contribution shows details on dust remobilization observations (direct one by the camera and indirect one via SEM comparisons), methods and instruments used for the experiment, and also introduces possible physical mechanisms responsible for the observed collective grain transport (unipolar arcs, eddy currents).
P3.127

**Initial definition of load conditions in DEMO**

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An essential goal of the EU fusion roadmap is the development of design and technology of a Demonstration Fusion Power Reactor (DEMO) to follow ITER. A pragmatic approach is advocated considering a pulsed tokamak based on mature technologies and reliable regimes of operation, extrapolated as far as possible from the ITER experience. The EUROfusion Power Plant Physics and Technology Department (PPPT) started the conceptual design of DEMO in 2014. This article will describe the most important load combinations that have to be considered in the design of the DEMO tokamak systems including their categorization into four classes based on the expected frequency of occurrence. Furthermore, with exception of the heat loads from the plasma particles and radiation to the plasma facing components, the most important load cases will be described, critical loads will be quantified, and their dependency on the tokamak design choices will be highlighted. These will include (i) dead weights, (ii) coolant operating pressures, (iii) electromagnetic (EM) loads during normal operation, (iv) EM loads due to toroidal field coil fast discharge, (v) EM loads in fast and slow plasma disruptions due to eddy and halo currents, (vi) neutron heat loads on plasma-facing components, in-vessel components and the vacuum vessel, (vii) loads in the dominant accident sequences identified by safety analysis.
Progress in EU-DEMO In-Vessel Components integration

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In the framework of the EUROfusion DEMO Programme, the Programme Management Unit (PMU) is assuming the role of the plant and tokamak design integration. It is recognized, in part thanks to the ITER experience, that due to the large number of complex systems assembled into the tokamak vessel for integration it is of vital importance to address the in-vessel integration at an early stage in the design process. Furthermore in DEMO the auxiliary, heating, and fueling systems integrated in the tokamak will have to interface with and be integrated into a breeding blanket and will face a harsh nuclear environment during operation. The in-vessel components as a whole will have to satisfy the top level requirements of remote maintainability and high reliability; however for the engineering integration of single systems inside the vessel and breeding blanket, a deep understanding of the requirements of the interfacing systems is mandatory and has to be developed at an early stage in the design process. In the EU DEMO design, after a first phase in which the different systems have been developed independently based on the baseline DEMO design defined by the PMU, an effort has been made here to define the interface requirements and to propose the integration strategies for the auxiliary, heating, and fueling systems into the vacuum vessel and the breeding blanket. This work presents the options studied, the engineering solutions proposed, and the issues highlighted for the in-vessel integration of the DEMO fueling lines, auxiliaries, heating systems, and diagnostics.
Identification of Blanket design points using an integrated multi-physics approach

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The development of the fusion technology reliability involves, among other issues, the improvement of simulation tools to be used for the design of reactor key components, such as the Breeding Blanket (BB), where the engineering requirements and constraints are of nuclear, material and safety kind. For this reason, advanced simulation tools are needed. In the European DEMO project, several efforts are currently dedicated to the development of an integrated simulation-design tool for DEMO BB that is able to carry out a multi-physics analysis of such a component, allowing the characterization of BB design points which are consistent from the neutronic, thermal-hydraulic and thermo-mechanic points of view. Within the framework of EUROfusion activities, a procedure, regarding the neutronics part, has been set-up to this end at Karlsruhe Institute of Technology (KIT). The first step of this approach requires the definition of the reference geometry, coming from generic CAD files and to be converted into more suitable formats for neutronic analysis with Monte Carlo codes such as MCNP5/6. In this study, the neutronic model of Helium Cooled Pebble Bed (HCPB) slice in the equatorial outboard module has been used for the characterization of BB design points. Therefore, the definition of proper boundary conditions has been pursued (i.e. angular distribution of neutron wall loading). For this reason, a 2D axial-symmetric neutronic model has been also developed to simulate all the reactor components that are radially and vertically based on reactor systems criteria and the materials are homogenized through the different regions. The present work aims to summarize the research activity carried out and results obtained are herewith reported and critically discussed. Furthermore, the strengths and weaknesses of this integrated coupling approach are highlighted and the potential developments, including the use of neutronic outcomes for CFD and thermo-mechanic analyses, are described as future steps.
Issues involved in the choice of low operating temperature for DEMO Eurofer divertor cassette

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Among the design activities of the DEMO divertor cassette carried out in the frame of EUROfusion an important parameter is the operating temperature of the divertor cassette. As for the DEMO breeding blanket Eurofer has been chosen as structural material of the divertor cassette due to its low long-term activation, low creep and swelling behavior under neutron fluence. The choice of the operating temperature, i.e. the coolant condition, is driven by different and often conflicting requirements and must consider a range of aspects, e.g. type and pressure of coolant, loss of coolant accident consequences, the application of design code, or the compatibility for power conversion in the balance of plant. Different options are therefore discussed in this article and the rationale for the selection is outlined. The first option aims at operating within the temperature window of 350-550°C that is recommended in order to reduce the degradation of the material properties due to the irradiation with high energy neutrons in DEMO. This aims in particular at avoiding a shift of the ductile-to-brittle-transition temperature to temperatures higher than room temperature that is known to occur at irradiation temperatures below ~350°C. This temperature level however practically excludes the use of liquid water as a coolant, leaving helium or steam as alternatives. Other options foresee operation at temperatures below 350°C using liquid water as coolant. The feasibility of such options is discussed focusing particularly on the impact of material degradation on the design assuming moderate neutron fluence as defined for the DEMO divertor cassette. In DEMO design it is currently foreseen that the divertor shall be replaced after two full power years. Neutronics analysis indicates that the maximum neutron damage in the Eurofer-based divertor cassette will reach only 6 dpa for the specified lifetime.
Systems engineering approach for pre-conceptual design of DEMO divertor cassette

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This paper presents the pre-conceptual design activities conducted for the European DEMO divertor, focusing on cassette design and Plasma Facing Components (PFC) integration. Following the systems engineering principles for the conceptual stage, high level design requirements are collected and conceptual 3D model of divertor’s cassette is presented. The work moved from the geometrical and interface constraints imposed by the 2015 DEMO configuration model. Then, since different materials will be used for cassette and PFCs, the divertor geometry has been developed taking into account the cooling parameters of the cassette Eurofer steel and the integration of PFCs cooling system. Accordingly, the design process led to a double wall cassette structure with internal reinforcing ribs to withstand cassette coolant pressure and three different kinds of piping schemes for PFCs with dual circuits. These three solutions differs in the feeding pipes layouts and target manifold protection and they have been proposed and evaluated considering heat flux issues, shielding problems, interface requirements with blanket and vacuum vessel and remote maintenance needs. Moreover, in this work two solutions for a two-body split cassette are proposed, with the aim of studying the feasibility of a separate inboard/outboard maintenance process. A cassette parametric shell model has been used to perform first structural analyses of the cassette body against coolant pressure. Taking advantages of the parametric surface modelling and its linkage with Finite Element (FE) code, the cassette ribs layout and thickness has been evaluated and optimized, considering at the same time the structural strength needed to withstand the coolant parameters and the maximum stiffness required for cassette preloading and locking needs.
Periodical replacement of in-vessel components is required for DEMO. The surface dose rate of in-vessel components for DEMO with fusion power of 1.5 GW is higher than that of shielding blanket in ITER by double digits. In addition, DEMO requires five-year cooling time for decreasing its dose rate to the level of ITER. Therefore, it is difficult to adopt the in-vessel maintenance scheme as ITER in terms of plant availability. To consider a maintenance process for DEMO, 3D shutdown dose rate map was analysed with MCNP-5 and DCHAIN-SP2001. The operation time of blanket (F82H) and divertor (W mono-block) is assumed to be 4 years and 1 year, respectively. The proposed maintenance scheme has an assumption that blanket integrated with shielding block (SB) is replaced through a vertical upper port whereas divertor also integrated with SB is replaced through a bottom port. To reduce dose rate, robot arm should approach from behind a SB and be fixed with an attachment of the SB. Based on the scheme, the dose rate in each maintenance port was evaluated. Before replacement, the spatial dose rate in a maintenance port for blanket and for divertor at the beginning of maintenance was 0.01 Gy/h and 0.1 Gy/h, respectively. When the divertor with the SB was removed, the spatial dose rate in each maintenance port was 100 Gy/h. After the replacement with new equipment, the dose rate in each maintenance port would be 10 Gy/h. The spatial dose rate in vacuum vessel of ITER during maintenance is determined as 250 Gy/h according to the requirement for remote handling of shielding blanket. The proposed maintenance scheme can limit the use of remote equipment only in a maintenance port leading to the reduction of the dose rate in the maintenance area for DEMO in comparison with ITER.
P3.134

Structural Assessments of the KDEMO Blanket Modules

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The Korean fusion demonstration reactor (K-DEMO) is in the early stages of conceptual design. Ceramic breeder blanket modules are being investigated. These have had extensive nuclear and thermal evaluations. Structural assessments are in process. This paper presents stress analyses performed at PPPL in support of the blanket design. Disruption loading, including the effects of ferromagnetic structural materials is evaluated. An approximate, but representative model of the blanket is used to evaluate a full set of normal thermal, pressure, and static magnetic loads. Disruption and faulted pressure loads are assessed as well. Inner and outer support shells are planned. The support shells serve as nuclear and electromagnetic shields for the vessel. This arrangement is a part of a vertical maintenance concept, that removes the inboard blanket module components with a radial and vertical traverse and leaves much of the massive shielding and support structure in place. Normal and Disruption blanket loads need to be quantified to show that these loads can be carried by the proposed structure, and to qualify the internals of the blanket modules. The KDEMO disruption analysis employs a simple modeling of the plasma by adjusting current densities in regions of the cross section defined for the plasma. The quench is modeled as a decay of the plasma current. Details of the blankets are developed from published descriptions of the KDEMO ceramic breeder concept. Disruption eddy current loading is quantified by imposing time dependent vector potential gradients from the simplified global disruption model on a more detailed representation of the blanket structure. The intention of this analysis is to develop tractable models of the blankets to investigate basic sizing and feasibility of the inboard and outboard blankets and their support mechanisms.
Rationale and Method for design of DEMO WCLL Breeding Blanket Poloidal Segmentation

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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). A new WCLL BB concept design has been proposed and investigated, starting from DEMO 2015 reference configuration. The first activity driving the BB design was the definition of the poloidal segmentation. Past studies were carried out to identify pro and cons of different approaches. Current trend in breeding blanket designs is based on the multi module box approach, which has advantages in terms of manufacturing; in reducing the global stress and strain during the start-up and the shut-down phases and during operation and in simplifying the First Wall (FW) layout and integration. Nevertheless, drawbacks are identified, such as the reduction of Tritium Breeding Ratio (TBR), the constraints in manifold and in Back Supporting Structure (BSS) design and integration because the limited space available. After a critical review of the rationale pursued in selecting the multi module box segmentation, the paper presents a method that defines and optimizes the main design drivers for WCLL BB segmentation. The method is based on the definition of Figures Of Merits (FOM), consisting in numerical parameters, such as the ratio between the modules volume and the overall volume of segment assigned, the approximation between the real profile of the modules and the theoretical one, the form factor of the modules, the ratio between the module thickness at the mid-plane and the segment thickness at the same position. The FOM support the choice among different options. In particular two different solutions of poloidal segmentation have been compared and, according to the proposed method, the best one was chosen for the design of WCLL BB.
Technological assessment between vertical and horizontal remote maintenance schemes for DEMO reactor

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Maintenance is one of the critical issues in the DEMO design. Several maintenance schemes has been comparatively evaluated from the viewpoint of plasma positional control, in-vessel transferring mechanism of blanket segment, and pipe connection in order to establish a feasible reactor maintenance scheme on the DEMO reactor. Two options has been selected as likely remote maintenance schemes on DEMO: the banana-shaped segment transport using all vertical maintenance ports (BSAV), and the saddle-shaped segment transport using limited number of horizontal maintenance ports (SSLH). First, those schemes are quantitatively examined from the viewpoints of space for welding/inspecting and access of maintenance devices, and we conclude that both schemes still have several technological issues, however, demonstration of both maintenance schemes in DEMO is considered as feasible. Maintenance scheme also affects the configuration of in-vessel components, the arrangement of poloidal field (PF) coils, which have an impact on the plasma performance/control. Especially, higher plasma elongation requires a passive stabilizing effect on the positional instability by the in-vessel components as well as the vacuum vessel. Hence, we evaluate the stabilizing effect depending on the maintenance schemes of BSAV and SSLH. We analyze the stabilizing effect by the eddy current on the in-vessel component and the vacuum vessel for both maintenance schemes of BSAV and SSLH. These results suggest that the conducting shell of SSLH has a higher stabilization effect on vertical stability than of BSAV. In contrast, comparing BSAV and SSLH on the control coil power, we find that the BSAV scheme reduces the control coil power. For SSLH, four control coils (PF coils) are located far from the equatorial plane, thus the control coil current in SSLH becomes greater than that in BSAV. Therefore, from an overall perspective, the BSAV scheme seems to have an advantage regarding plasma vertical stability.
Dynamic Model Identification Method of Manipulator for inside DEMO Engineering

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In the inside engineering of DEMO, the robotic machines or manipulators are foreseeable to be widely employed, which often have to deal with the demanding working conditions. The construction of the dynamic model of the robotic machine or manipulator can not only benefit the performance evaluation of the manipulator in the early design stage, but also can be incorporated into the control system of the robot or manipulator, in practical level, to gain the high control performance. However, in practice, it is rather difficult to construct accurately the analytical dynamic model for the robots or manipulators. The reasons behind include, but not limited to, lacking the physical insight of some dynamic phenomenon, the inaccuracy or infeasibility of the direct measurements or the deviation of some dynamic properties after the robot or manipulator’s assembly and deployment. A method of constructing the dynamic model of robot with the unknown parts is proposed. The method can identify the unknown parts of the dynamic system by incorporating a BP neural network that will substitute the unknown parts in the system after the well training. A modified Levenberg-Marquardt algorithm is developed for the training of BP neural network, which can back propagate the errors between entire actual system and the constructed model into the training process of neural network. For general application, an example of constructing the dynamic model for a general second order mechanic system with unknown dynamic component is presented. For the further validation on the complicated structure, the method is applied to a 10 DOF robotic machine. The friction models in the robot are taken as the unknown dynamic parts. After incorporating the BP neural network, the dynamic model of the entire robotic machine are successfully established. The proposed dynamic model identification method can also be applied to a general case.
FFMECA and recovery strategies for ex-vessel remote maintenance systems in DEMO

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In DEMO, the ex-vessel Remote Maintenance Systems (RMS) are responsible for the replacement and transportation of the plasma facing components. The ex-vessel operations of transportation are performed by cranes or by means of cask transfer systems (CTS) moved by trolleys. The main loads of transportation are the blankets and divertors. The blankets are extracted and transported vertically by cranes along galleries from the reactor to the storage or maintenance areas. An alternative is the transportation in horizontal configuration by means of CTS system along the galleries. The divertors are transported by means of trolleys. A failure may occur in any situation, interrupting the current nominal operation. The work identifies a functional breakdown structure for the ex-vessel RMS operations and develops a Functional Failure Modes, Effects and Criticality Analysis (FFMECA). The results of the different FFMECA studies lead to the conclusions in terms of the most critical failure scenarios and the pros and cons of the horizontal versus the vertical transportation of blankets. In case of failure, a recovery procedure shall be triggered. A recovery strategy is presented according to the following priorities: i) resuming the current nominal operation, ii) perform a different nominal operation and iii) perform a rescue operation. For radiation protection aspects, the recovery strategy is divided in two parts: with activated components (when transporting or in the proximity of activated components) and without activated components (free of load or transporting non activated components). The results will help the design process to improve and thus reduce the criticality index of the identified failures. 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.
Impact on remote maintenance of varying aspect ratio and TF coil quantity for DEMO

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As part of the conceptual design studies for a European DEMO, a range of Tokamak geometries are being considered. As identified in the EFDA Roadmap to the realisation of Fusion Energy: “The integration of the Remote Maintenance system within the DEMO plant is an essential task within the DEMO CDA phase. This will involve establishing requirements, functions and interfaces with many other systems to ensure that plant availability and maintainability are considered from the outset.” In order to fulfil this integration requirement these different geometries of DEMO have been assessed for their impact on Remote Maintenance (RM). The aspect ratio and number of TF coils have been identified as the pivotal variables, driving the Tokamak geometry, with significant effects on Remote Maintenance both in terms of technical feasibility and speed of operation. This paper will quantify the effect of varying the number of TF coils and the aspect ratio of a European DEMO on its RM system, in terms of technical feasibility and speed of Remote Maintenance operations. It will compare Tokamak geometries with aspect ratios from 2.6 to 4.0 and 16 or 18 TF coils. The different geometries will be compared using key performance related parameters, with appropriate weightings applied, as defined by RM experts from RACE. The parameters focus on the removal and replacement of the Blanket Multi-Module Segments (MMS) through the upper vertical ports, as this is deemed to be one of the main drivers for the speed and technical feasibility of the RM in DEMO.
Reliability prediction method for DEMO divertor remote maintenance concepts using stochastic Petri nets

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The next European fusion reactor after ITER is called DEMO. The development implementing ITER experiences has taken place within EUROfusion Programme. One of the reactor maintenance system development tasks has been focused on Divertor Maintenance system. The maintenance of DEMO involving handling hazardous components shall be carried out remotely such as the installation and removal of the Divertor area. In order to perform the installation and removal of the critical in-vessel components of DEMO, dedicated RH devices shall be developed to carry out remote operations. Plant availability is one of the top level requirements for DEMO, therefore maintenance reliability and effectivity are critical issues. This paper suggests an innovative method for quantitatively assessing the reliability of RH concept design. The method described in this paper aims to support the engineers during the design process for quantitatively assessing different design options based on a predictive reliability approach. High level operational sequence of the RH operation is modelled as a stochastic Petri Net. Reliability of different concept is obtained and compared to determine which concept is more reliable regarding a particular task, subtask or the entire remote handling sequence. Finally the paper presents the quantitative results of the different remote handling concept designs for two potential DEMO maintenance port configurations. Benefits of such method used during the conceptual design phase of complex systems such as in DEMO remote maintenance operations are mentioned and further development of such method are also discussed in the conclusion of this paper. The work behind this paper takes place in the DEMO Remote Maintenance Project activities implemented under the EUROFusion Consortium. The purpose of the work package concentrating on divertor maintenance was to study and develop technical solutions for Divertor RM and perform concept evaluation based on remote maintenance point of view.
ADITYA Upgrade Vacuum Vessel: Design, Construction, Testing, Installation and Operation

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The First Indian tokamak, ADITYA had successfully completed 25 years of operation of limiter plasma at the Institute for Plasma Research (IPR). After achieving the targeted plasma and successfully carrying out many major tokamak experiments, the up-gradation of ADITYA tokamak with diverter configuration was planned. The upgradation includes the replacement of rectangular cross section vacuum vessel by circular cross section vacuum vessel to accommodate divertor coils in space between vessel and toroidal field coils. The new toroidal vacuum vessel has been designed with two semi-tori having electrical isolation at two junctions. The major radius (0.75 m) and minor radius (0.25 m) has been kept same as of old torus so as to accommodate the new torus inside the toroidal field coils. In order to accommodate as many numbers of diagnostics as possible, the ADITYA-U vessel is designed to have 112 port openings compared to 48 ports in ADITYA vessel. The leak proof, UHV condition, precise dimensions and lots of weld and demountable joints in the new vessel made the fabrication job very challenging. The ADITYA-U vacuum vessel of SS304L has been precisely fabricated as per IPR design by Godrej & Boyce Mfg. Co. Ltd. under the supervision of IPR scientists. The factory and IPR site acceptance tests of new vessel have been completed successfully, such as dimensional measurement, leak test, pressure, baking. The final acceptable tests have been carried out successfully as results of local leak rate <5x10^{-10} mbar.l/s and global leak rate <5x10^{-8} mbar.l/s, UHV test as <10^{-8}-9 mbar, vessel baking >150°C. After external testing, the vessel is installed successfully in ADITYA-U tokamak structure. In this paper, we present the all details of ADITYA-U vacuum vessel like design, analysis, fabrication, acceptance, installation and operation to ascertain its suitability for the plasma experiments.
Design optimization of structural components for the helical fusion reactor FFHR-d1 with challenging options

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The design activity of a conceptual design of a helical fusion reactor FFHR-d1 is progressing at the National Institute for Fusion Science. The superconducting magnet system of FFHR-d1 comprises one pair of helical coils, two sets of vertical field coils, and the coil support structure. The major and the minor radii of the helical coil are 5.6 m and 3.774 m, respectively. The magnetic field at the plasma center is 4.7 T. The coil support structure is designed from the perspective of the allowable stress of the material. Thus, it has apertures that are as large as possible to maintain in-vessel components. A continuous helical coil winding with a low temperature superconductor and a divertor made of tungsten and copper alloy with water cooling are considered for use in the reactor. These specifications comprised the basic option of FFHR-d1. In addition, there are several flexible design options. These options adopt new ideas that can solve certain issues in the basic option. The issues with the basic option include the winding method of the huge structure, high heat flux and neutron irradiation on the divertor, and the narrow radial build clearance. These alternative design proposals are treated as the challenging options. For example, a joint coil winding with a high temperature superconductor, a liquid metal divertor with molten tin, and an additional helical coil with a negative current flow that widens the distance between the plasma surface and the main helical coil are proposed. These proposals can be implemented by modifying the structural components of the basic option. In addition, design optimizations will be conducted by considering factors including mechanical soundness, magnetic field precision because of the deformation of the coils, an assembling/maintenance procedure, and a reduction in total weight via a finite element analysis and 3D model printing.
Review of Hydrogen Isotopes Transport Parameter

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The hydrogen isotopes Tritium and Deuterium will be the fuel of future fusion power plants. These isotopes will be in contact with components of the reactor, as well as with auxiliary systems. For safety studies and the overall Tritium budget, hydrogen transport parameters are necessary to perform according analyses. Reduced Activating Ferritic Martensitic (RAFM) steels at operation conditions with varying temperature between 300°C and 550°C can store and transport Hydrogen to a considerable extent. Also the knowledge of the transport parameters of austenitic steels will be needed for materials of pipelines and equipment in a power plant. Therefore, a qualified data base will be necessary for the future work. Our contribution presents firstly an overview of transport parameters of different RAFM steels. Secondly a short review of austenitic steels for tubes and tanks, also regarding exothermic hydride generating materials. As third part the publication will review the transport parameter of elementary materials, generating understanding of the first both parts. The authors are preparing experiments for the determination hydrogen transport parameter of elevated temperature. This work has been carried out within the framework of the EUROfuson Consortium, WPPFC, and has received funding from the Euratom research and training program 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.
Tritium permeation through containment structures is an important factor for safety and design analysis of fusion energy systems. This process controls several key aspects of the system performance, including the amount of radioactive tritium released to environment, the requirements on tritium breeding ratio, the tritium recycling from the first wall, and it influences the selection of technologies adopted in the design of several components. As a consequence the characterization of permeation and diffusion process and the determination of relevant material properties is a priority for the design of breeding blanket systems. Considerable data is available for hydrogen isotope diffusion and permeation in different metals as a function of temperature. However, in the case of breeding blankets, these materials will be additionally subjected to a temperature gradient between the inner and outer surfaces of the coolant circulating channels. Although design analysis has shown that this factor could have an important impact on the system performance [1,2], permeation driven by thermal gradients (also referred to as Soret effect) has not in general been included in either experimental studies, or modelling of blanket systems, and must be assessed. The THERMO-PERM facility has been developed for this purpose as an external extension to the CIEMAT Van de Graaff electron accelerator laboratory. This paper describes the experimental system, in particular the new permeation chamber whose cooling/heating system allows one to produce thermal gradients up to about 120 °C in 1 mm thick samples and measure hydrogen isotope (H2 and D2) permeation. Experimental results obtained for deuterium in 316 L stainless steel are also discussed. [1] O. V. Ogorodnikova, et al, Journal of Nuclear Materials 273 (1999) 66-78. [2] G. R. Longhurst, Journal of Nuclear Materials 131 (1985) 61-69.
Al2O3 coatings as barrier against corrosion in Pb-Li

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Al-based coatings are proposed as anti-permeation and anti-corrosion barrier in Pb-Li breeding blankets - Water Cooled Lithium-Lead (WCLL), Helium Cooled Lithium-Lead (HCLL) and Dual Coolant Lithium-Lead (DCLL). In this work, Al2O3 coatings have been prepared by Pulsed Laser Deposition (PLD) at Istituto Italiano di Tecnologia (IIT) and they have been qualified in Pb-Li to evaluate its suitability from the point of view of corrosion. Some samples of EUROFER, as-received and coated with Al2O3, have been exposed in Pb-17Li, at ENEA, in stagnant conditions for 1000 hours at 550 °C, which is the maximum operating temperature expected in the three models of Pb-Li blanket. Full characterization of the samples, before and after the corrosion tests, has been carried out using different techniques such as attenuated total reflectance spectroscopy (IR-ATR), scanning electron microscopy (SEM), energy dispersive x-ray fluorescence (EDX), confocal microscopy and secondary ion mass spectrometry (SIMS) to evaluate the migration of Li into the EUROFER. The surfaces of the uncoated samples are damaged irregularly due to the dissolution of the steel into the liquid metal. The corrosion process, observed only in the uncoated specimens, produces chromium desegregation. Thus, the EUROFER steel becomes poor in chromium in the region which has been in contact with the liquid metal. Furthermore, Li penetration into EUROFER has been observed by SIMS. By contrast, the corrosion process is negligible for Al2O3 coated samples. On these surfaces, it is observed that Li and Pb go through the alumina coating forming a thick interface between the coating and the steel. These results suggest a good performance of the alumina coatings under the test conditions.
Deuterium permeation behavior in iron-irradiated erbium oxide coating

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Tritium permeation through structural materials in fusion blankets is one of the most important issues in terms of a fuel loss and radiological hazard. Tritium permeation barriers (TPBs) have been developed for several decades, and erbium oxide (Er\textsubscript{2}O\textsubscript{3}) coatings have recently been intensively studied as TPBs. However, irradiation effects in TPB coatings on hydrogen isotope permeation have not been elucidated in spite of a severe radiation environment in the blanket region. In this study, the correlation between irradiation damages generated by iron ion at elevated temperature and deuterium permeation in the Er\textsubscript{2}O\textsubscript{3} coatings has been investigated. Er\textsubscript{2}O\textsubscript{3} coatings were fabricated by filtered vacuum arc deposition on reduced activation ferritic/martensitic steel substrates. Iron irradiation on the coated samples has been conducted using a tandem accelerator DuET at Kyoto University. The ion energy and the irradiation temperature were 6.4 MeV of Fe\textsuperscript{3+} at 873 K. The average displacement was 0.01\textsuperscript{1.0} dpa, and the damage was distributed in the whole thickness of the coatings (1 \textmu m) according to a calculation. Finally, deuterium permeation measurements have been performed using a gas-driven permeation system. Deuterium permeability of the coated samples irradiated to 0.1 dpa at 873 K was 75\textsuperscript{80\%} lower than that of unirradiated one, and decreased by a factor of up to 3200 in comparison with that of an uncoated substrate. In addition, the activation energy of permeation was estimated to be 91 kJ mol\textsuperscript{1}, which was clearly higher than that of the unirradiated coating (60 kJ mol\textsuperscript{1}). These results indicate that the irradiation under elevated temperature contributes to the modification of the grain structure of the coating by irradiation damage introduction and recovery. Irradiation temperature and damage concentration dependences will be also discussed in the presentation.
Compatibility test between ceramic breeder and EUROFER steel for European HCPB-DEMO blanket design

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In a helium cooled pebble bed (HCPB) DEMO reactor, ceramic breeder pebbles are packed in EUROFER structural steel blanket and generate tritium as a consequence of the reactions between lithium and neutrons. As breeder pebbles and EUROFER are contacted at a high temperature for a long period during the operation, corrosive attack to EUROFER could occur even with the low activities of ceramic breeder materials. Additionally, traces of metals diffused from EUROFER into breeder pebbles could also have critical influence on activation property, resulting in a longer storage period for breeder recycling possibly. This study aims to investigate the compatibility between breeder material (Li4SiO4 with added Li2TiO3) and EUROFER steel under the blanket atmosphere by investigating corrosion layer near the surfaces. In an atmosphere controllable tube furnace, alumina crucibles where a breeder pellet made from breeder pebbles is sandwiched by two EUROFER plates by using force from an Inconel spring, were heated under He with 0.1% H2 atmosphere for up to 1 month at both elevated temperatures and those expected within the blanket. The results of surface XRD showed that the surfaces of EUROFER plates were partially oxidized, even in the reducing condition, and lithiated due to Li diffusion from the breeder pellet. On the other hand, diffusion of metals from EUROFER pellet was observed on the surface of breeder pellets. Effective diffusion coefficients estimated by measuring the thickness of corrosion layer after several heating periods will be also presented in the poster presentation in order to estimate possible diffusion length of the corrosion layer after the use under the blanket condition.
Development of electron beam welding technologies for fabrication of the KO HCCR TBM

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Korea has designed a helium cooled ceramic reflector (HCCR) test blanket module (TBM), including a TBM shield, called a TBM set, that will be tested in ITER. The HCCR TBM is composed of four sub-modules and a back manifold. In addition, each sub-module is composed of a first wall (FW), a breeding box with seven-layer breeding zone (BZ), and side walls with the cooling path. Korean RAFM steel was developed as a structural material for the HCCR TBM, and advanced reduced activation alloy (ARAA) was selected as the primary candidate from various program alloys. Welding technologies for fabrication of the HCCR TBM were developed using ARAA. To establish and optimize welding procedures for electron beam welding of an ARAA material, variations in welding current and speed were investigated. Tensile, hardness, impact, bend, and microstructure characteristics were performed before and after post-weld heat treatment to evaluate the welded specimen under the determined welding conditions.
Design and fabrication of a Permeator Against Vacuum prototype for testing at Lead-Lithium facility

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Tritium recovery is one of the major issues of a future DEMO reactor, in order to accomplish with the requirement of tritium self-sufficiency. Different techniques have been proposed over the years for the extraction of tritium, depending on the Breeding Blanket technology. After a preliminary selection, the EUROfusion Programme has considered the Permeation Against Vacuum (PAV) technique as baseline for those blankets which use PbLi as breeder. In this framework, an experimental PbLi loop is under construction at CIEMAT with the main objective of testing the PAV technique operating at conditions relevant for the actual European DCLL breeding blanket (i.e. high PbLi velocity, maximum temperature of 550 °C, low tritium concentration). A conceptual design of a squared multi-channel PAV for its implementation in the loop has been produced. Its geometrical and physical characteristics have been obtained through a detailed analysis of the efficiency, defined as the difference between tritium concentrations at the entrance and exit of the extraction unit. With regard to the PAV membrane, previous studies have shown the feasibility of using tantalum or niobium due to their good permeability and compatibility with PbLi. However, in order to save costs and time a preliminary α-Fe membrane has been considered for the prototype, in spite of its low permeability. Notwithstanding this first device is considered a low-efficiency prototype it is able to demonstrate the feasibility of the technique. Structural calculations are also presented and evaluated for the development of the best implementation of the manufacturing process, paying special attention to the interface between the membranes and the main structure in order to avoid PbLi leakages. Other important aspects, such the preservation of an adequate vacuum level and its arrangement system, have been also considered.
P3.152

**Modeling blanket ferromagnetic loading using edge potential elements**

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Modeling Blanket Ferromagnetic Loading using Edge Potential Elements

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Future fusion experiments and reactors will require first wall materials that can survive the thermal and nuclear radiation environment without structural degradation. Candidate materials that are under consideration include Reduced Activation Ferritic Martensitic (RAFM) steels such as Eurofer 97 and F82H. These materials are ferromagnetic and will alter the magnetic fields that not only effect the plasma confinement but result in additional forces on the structural components which must be understood, quantified and factored into the design. These magnetic forces are in addition to the eddy current Lorentz forces that result during a plasma disruption. They are present during normal steady state operation but will also result in higher flux swings within components during a disruption driving eddy currents and Lorentz forces even higher. This paper investigates the loads resulting from the use of RAFM as a blanket material. The ANSYS code Edge Potential Elements are used to solve the non-linear transient electromagnetic field problem. The edge potential method overcomes some of the flux discontinuity issues related to ferromagnetic materials as compared to the magnetic vector potential formulation. It has been applied to ITER scale machines where the RAfm will reach magnetic saturation. Results are compared to simplified methods that may be useful for preliminary design studies. Poster presentation preferred.
Sensitivity analysis of tritium retention in tungsten walls using a fusion reactor simulation code

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For the fusion reactor operations, the tritium (T) retention and permeation in the reactor walls are important for points of views of safety and fuel cycle. It is known that T retention in tungsten (W) is less severe compared with carbon (C). However, recent experimental studies revealed that the neutron irradiated damage, surface recrystallization, and fuzz formation by He ion irradiation increases the retention amount. It was also revealed that the T desorption flux strongly depends on the surface temperature because of the difference of potential energy of the T trapping sites. It is expected that desorption occur not only during the plasma operation but also the intervals. Thus, a comprehensive study for the operational scenario should be taken in order to estimate the T retention in a W wall device. In this study, a T retention model was constructed and integrated into a reactor operation simulation code. In the code, the main plasma was treated in a 0D model and the SOL and divertor plasma was treated in the two-points model or 1D fluid model. For the wall, 1D heat conduction for the depth direction and erosion by sputtering and evaporations were solved for each plasma facing components (PFCs). By taking sequential calculation with the nuclear calculation codes, effects of the nuclear heating and the neutron damage can be included with considerations of the poloidal asymmetry. Based on the experimental data, T retention and desorption amounts are modeled as a function of surface temperature. Using the operation simulation code with the retention model, sensitivities of parameters, plasma operation intervals, design of PFCs, effects of the permeation barrier and neutron damages, for the T retention amount are evaluated.
Out-of-pile tritium adsorption/release behavior of advanced EU breeder pebbles

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Advanced ceramic breeder pebbles composed of a mixture of Li4SiO4 (LOS) and Li2TiO3 (LMT) are fabricated and developed at KIT by a melt-based process (KALOS). The produced pebbles are easily characterized for their non-nuclear properties. Yet, as the main properties of a tritium breeder material are the generation and release of tritium, these characteristics also have to be examined. Neutron irradiation experiments are expensive and require several years of planning, preparation and post irradiation examination in addition to several years of irradiation. Therefore, the comparably rapid development of the breeder pebbles cannot be supported by classical neutron irradiation experiments. Out-of-pile loading of pebbles with tritium can be conducted much more easily and frequently, while providing experimental data of avail in this regard. Within this work, firstly, the tritium loading of current grades of advanced ceramic breeder pebbles with three different LOS/LMT compositions (20-30 mol% LMT in LOS) and pebbles of EU reference material, was performed in a consistent way. Secondly, the controlled, temperature dependent release of the introduced tritium was investigated by temperature programmed desorption (TPD) experiments to gain insight into the desorption characteristics. As the pebble size can be an influential parameter on the measurements, this factor was addressed by using monosized spheres. The obtained TPD data was decomposed into individual release mechanisms according to well-established desorption kinetics. The analysis showed that neither the pebble diameter nor the pebble composition of the tested samples severely change the release behavior. Yet, an increased content of lithium metatitanate leads to additional desorption peaks at medium temperatures. The majority of tritium is released by high temperature release mechanisms of chemisorbed tritium, while the release of physisorbed tritium is marginal in comparison. The results allow valuable projections for the tritium release behavior in a fusion blanket.
Study on pressure resistance and separation of hydrogen isotopes of Palladium packed columns

LIU, Zhenxing

Abstract: This paper presents the results of experimental study of the columns packed with Palladium deposited on kieselguhr (Pd/k). The characteristic of pressure resistance and separation of hydrogen isotopes of the Pd/k column was investigated. The corresponding relationships among pressure resistance characteristics of Pd/k separation column and Pd/k material physicochemical properties, column structure, and the filling coefficient were determined. The relation curve between the pressure drop per unit length and the velocity was established under different inner diameter. And the analysis model between pressure resistance and velocity was set up by calculating the pressure resistance characteristics parameters under certain condition. The separation of hydrogen isotopes (H, D) is performed by displacement chromatography on palladium packed column. The results showed that the Van Deemter model gives a fairly good analysis of the process. Under the experimental condition, the minimum of height equivalent to a theoretical plate (HETP) is less than 0.64cm, which shows the high separation efficiency of packed column for hydrogen isotope separation. The results indicate that the Pd/k column can realize the separation of hydrogen isotopes. This hydrogen isotope processing can apply to the separation of hydrogen isotope gases of the D/T fuel cycle system for fusion reactor.
Quantifying TBR uncertainty due to nuclear data in fusion blanket modelling

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Tritium self-sufficiency is a fundamental requirement for future DT fusion demonstration and commercial power plants. Hence, prior to the construction of expensive, complex fusion breeder blanket assemblies there should be a concerted effort to quantify and ultimately reduce the uncertainties associated with various nuclear observables. This will enable tritium self-sufficient blankets to be designed with a high degree of confidence. The largest source of uncertainty for tritium breeding analysis is poor nuclear data. ‘Clean’ experimental work to better characterise individual fusion relevant isotopes has been undertaken at the University of Osaka and JAERI in Japan. ‘Design-specific’ experiments have been conducted at FNG, Italy where a blanket mock-up was irradiated and tritium production rates were measured. The work presented here employs a recently developed simulation methodology, Total Monte Carlo (TMC), applied to the problem of uncertainty due to nuclear data. TMC uses nuclear data files produced by nuclear physics models. By changing values of fundamental nuclear parameters input to the T6 code system, nuclear data is generated for given isotopes. The neutronics of a problem is then simulated many times with this varied data, making it possible to produce PDFs of the desired observable quantities. In the work presented, the nuclear data uncertainty is quantified for observables, such as the Tritium Breeding Ratio (TBR) and neutron multiplication factor, for a lithium lead fusion blanket design.
Neutron activation of impurity seeding gases within a DEMO environment

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In preparation for the design of a future tritium-handling plant for the DEMO fusion reactor, a study was undertaken to consider the activation of gases, in addition to those used as fuel, which are to be injected into DEMO for the purpose of reducing damage to the divertor. Likely impurity gases were identified as nitrogen, neon, argon, krypton and xenon, with no clear consensus as to which were preferred, and in what quantities these would be injected. Modelling with the FISPACT-II code, using the TENDL-2014 libraries as input, was performed to determine the products arising from their activation by the neutron flux expected at the DEMO first wall. The subsequent evolution of the progeny was investigated, along with some variation in the activation time. Results indicate a variety of progeny are produced, most of which are short lived. However, the gamma emission from some progeny such as argon-41 from argon injection, from some metastable states and other isotopes of krypton and bromine following krypton injection, and from some metastable states of xenon following injection of xenon, could potentially cause concern. The possibility of the creation of hydrogen fluoride and nitric acid also exists within the DEMO and tritium plant systems. This work has been carried out within the framework of the EUROfusion Consortium under the Tritium, Matter Injection and Vacuum programme (WP TFV).
Looking towards the future of fusion devices, detailed understanding of the underlying working properties is desired knowledge. Even though there are many fusion devices available and extensive operating data is being collected, computational analysis is an underlying requirement to fully understand how a fusion device will operate. Due to the extensive complexity of fusion devices a computational method must be devised that will allow for quick computational iterations while minimizing the error due to the complexity of the calculation. Large amounts of data can be computed on the structural, magnetic, nuclear, etc. and all of this information is needed to develop a true picture of how a certain environment will react. Each aspect of this information is calculated independent of each other. This can cause extensive delays when trying to build the broader picture of a fusion device because of the use of separate solution methods that require different inputs. The nuclear analysis aspect is especially important, and much of the information found through nuclear analysis must be transferred over to other analyses. At PPPL we have developed a method to move this information from the nuclear analysis done using the Attila neutronics code to other analyses using different codes such as ANSYS. A simple single breeding blanket was used to help develop this method. The setup of a single blanket allows for quick iterations to adjust the blanket parameters while maintaining the fundamental nuclear responses. Once decisions are made from the single blanket, then the model can be scaled up to incorporate the different device shapes that will affect the overall nuclear response. This report will focus on the nuclear calculations, the results of these calculations, and the transfer of the data to different analysis programs.
Coupled neutronic and engineering analysis of the Helium Cooled Pebbled Bed with Be12Ti

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The Helium Cooled Pebble Bed (HCPB) breeder blanket is being developed as part of the European Fusion Programme. Part of the programme is to investigate blanket modules relevant for future demonstration fusion power plants. This paper presents fluid dynamic, thermomechanical and neutronic analyses of the helium cooled pebble bed with an alternative neutron multiplier, Be12Ti which is incorporated into the design replacing current Be multiplier. Potential benefits of switching to Be12Ti are reduced swelling and higher temperature limits. Alterations to the MCNP blanket geometry model were made to optimise the tritium production with the new multiplier. Neutronics simulations were performed using MCNP 6.1.1 unstructured mesh geometry models to find volumetric heat loads. Heat load were subsequently used as inputs directly on to a finite element in order to perform multiphysics engineering analysis. This paper provides insight, supported by neutronic and engineering analysis, on the validity of the design and comments on some of the potential advantages and disadvantages of using Be12Ti in the HCPB. Specific areas covered include tritium production, heat generation, material temperature limits and stresses.
The investigation of Ceramic Breeders (CB) is of great concern for the development of the solid breeder concept for ITER and DEMO. 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 irradiation as well as thermomechanical stresses have to be fulfilled. Lithium orthosilicate (Li4SiO4), applied as pebbles, has been chosen as reference material in the European Helium Cooled Pebble Bed (HCPB) due to its favourable properties. Studying the behaviour of CB pebbles exposed to relevant neutron irradiation is very important for the development of the EU HCPB and a rare opportunity. Therefore, the HICU experiment (high neutron fluence irradiation of pebble stacks for fusion) was carried out in the High Flux Reactor (HFR) in Petten (Netherlands). Different grades of Li4SiO4 pebbles containing a surplus of 2.5 wt.% SiO2 and different 6Li-contents up to 20 wt.% were included in the irradiation under DEMO relevant conditions (20-25 dpa, 400-900 °C). Selected results of the Post-Irradiation Examination (PIE) on Li4SiO4 samples will be presented with a focus on the pebble stability and morphology. Changes in the surface morphology, the microstructure and the porosity will be identified by optical microscopy/SEM and Archimedean immersion/He pycnometry, respectively. Investigations of the mechanical strength using crush load tests will reveal possible deteriorations due to neutron irradiation. The presented results will demonstrate the state of development of CB pebbles and will significantly contribute to the knowledge of CB pebbles' properties in a fusion relevant environment.
Post irradiation examination of High Dose irradiated Beryllium (HIDOBE-02) in the High Flux Reactor

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In the frame of the European Tritium Breeder blanket development for DEMO, two high dose irradiations of beryllium and beryllides, to be used as neutron multiplier, have been performed in the High Flux Reactor Petten (NL). From one irradiation, to 3000 appm He, the post irradiation results have been published in previous proceedings. In the second High Dose Beryllium irradiation (HIDOBE-02), various grades of beryllium pebbles, pellets and Titanium Beryllide samples have been neutron irradiated in the temperature range of 425-850 °C until the 30% End of Life (EOL) DEMO relevant Helium content of around 6000 appm He was achieved. The Post-Irradiation Examination (PIE) of HIDOBE samples has the objective to collect and analyse information about changes in the Be materials properties after irradiation. Measurements include dimensional changes, microstructure, thermal properties and tritium release. These results are compared with as received material and Beryllium irradiated to 3000 appm He. This paper will focus on the microstructure of the unconstrained and contrained pebbles and beryllide grades. It will be shown that after irradiation the material shows open porosity, which is dependent on irradiation temperature. Also the correlation of the irradiation temperature, porosity and results from tritium release desorption experiments will be investigated.
Effects of temperature and hydrogen pressure on the activation behavior of ZrCo

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Fast and efficient activation of ZrCo is beneficial to promote its application to hydrogen isotopes storage in the fusion energy field. To obtain the optimum activation procedures, the influences of temperature and hydrogen pressure on the activation behavior of ZrCo were systematically investigated. Experimental results showed that fast and efficient activation of ZrCo could be achieved by optimizing the temperature and hydrogen pressure conditions of the activation procedures including initial evacuation, hydrogenation and dehydrogenation. It was found that initial evacuation at temperature higher than 300 °C was clearly beneficial to the subsequent hydrogenation process. Hydrogen absorption rate of ZrCo during the activation process could be enhanced by increasing the hydrogenation temperature, whereas hydrogen pressures had an indiscernible impact on the hydrogenation process. Compared with other temperatures and hydrogen pressures, 100 °C and 0.8 bar H\textsubscript{2} was a preferred condition for hydrogenation of ZrCo. In addition, it was demonstrated that dehydrogenation at high temperature over 500 °C was favorable to enable activated ZrCo own high hydrogen capacity. As a result, optimum procedures composed of initial evacuation at 500 °C, hydrogenation at 100 °C under 0.8 bar H\textsubscript{2} and dehydrogenation at 500 °C under vacuum was highly recommended for fast and efficient activation of ZrCo.
P3.163

The assessment of shutdown dose rate of HCSB during its replacement in CFETR

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Chinese Fusion Engineering Test Reactor [Chinese Fusion Engineering Test Reactor] CFETR is a necessary and feasible engineering test reactor which aims at developing the fusion energy while the helium cooled solid breeder blanket (HCSB) is one of the most significant component of it. During the reactor operation stage, the blanket will be activated to produce highly radioactive substances by high energy neutrons irradiation. In order to protect the workers from over-proof radiation and ensure the normal operation of machinery, it’s necessary to evaluate the shutdown dose rate of HCSB during its replacement. According to the rigorous two-step (R2S) method, a shutdown dose rate calculation code was developed for nuclear devices with large dimension and complex geometries. The code integrated the functions of neutron transport calculation, activation calculation and decay gamma transport calculation by coupling the Monte Carlo particle transport calculation code MCNP with the European activation simulation code FISPACT. Based on the present design of HCSB, the radiation dose assessment of the HCSB during its replacement has been calculated and analyzed to verify the radiation safety of the present design. Key Words: Rigorous 2-step method; CFETR; HCSB; Radiation dose rate
Decay heat produced by neutron irradiation can lead to temperature rise in blanket even after plasma shutdown. The excessive temperature increase of blanket structure would be concerned with increase of decay heat when assuming loss of cooling capability for blanket even though vacuum vessel is assumed to be normally cooled with a safety function. The neutron wall loading is designed to be 0.78 MW/m² in ITER and become larger in DEMO. Thus, mitigating temperature rise of DEMO blanket caused by decay heat should be examined for integrity of in-vessel components. We focused on passive cooling performance of blanket structure with ribs under the assumption that coolant in the blanket was totally lost. Arrangement of ribs, direction of cooling channels in the ribs, and width of the rib were considered as design parameters. For the different models decay heat with neutron wall loading up to 3MW/m² was calculated by using a 2D nuclear-thermal-coupled analysis code. The obtained decay heat was applied to finite element model as boundary condition. Radiation condition was considered to a back wall of the blanket. Insulation condition was applied to the other faces of the blanket box. The thermal response of the blanket with different configuration was evaluated by using FEM codes. The maximum temperature of the blanket was different according to arrangement of the ribs. The temperature of blanket with ribs in parallel to the first wall was higher than that of the blanket placing the ribs vertically to the first wall. The temperature difference was over 400 K. In addition natural convection of helium gas injected in vacuum vessel was considered. The thermal hydraulic analysis was conducted to investigate thermal and hydrodynamic characteristics of helium gas in the vacuum vessel. Injection of helium gas contributed to mitigating temperature increase of the blanket.
Lithium-containing ceramics (Li-ceramics) are considered as tritium breeding material in pebble-bed form for solid-type breeding blanket in fusion reactor. The tritium breeding material requires small particle size to reduce diffusion distance of generated tritium in the intercrystalline. In addition, the essential resource, especially enriched Li-6, has to recover from the used tritium breeding material. For the hands-on operation during the recovery process, activation level of the used breeder is strictly limited in order to reduce impurities of the long-lived radioactive nuclides in the tritium breeding material. This study aims at the fabrication of Li2TiO3 pebbles, which have a small particle size and low levels of the long-lived radioactive nuclides, by using solid-state reaction process. Lithium oxide (Li2O) and titanium dioxide (TiO2) were used as starting materials in the synthesis of Li2TiO3 powder. First, the starting materials mixed by wet ball-mill process. The mixed powder synthesized by heat treatment. The crystalline structures of the synthesized powders were identified by X-ray diffraction (XRD) method. The elemental concentration of the synthesized powder was analyzed to investigate impurities by inductively coupled plasma-optical emission spectroscopy (ICP-OES). Scanning electron microscopy (SEM) was used to observe the size and shape of the synthesized powders. The average particle size of synthesized Li2TiO3 powder was approximately 150 nm. And, the long-lived radioactive nuclides, such as Aluminum (Al) and Cobalt (Co), were not flowed into the synthesized Li2TiO3 powder. The Li2TiO3 pebbles were fabricated by slurry droplet wetting method using the synthesized Li2TiO3 nano-powder. The characteristics of fabricated Li2TiO3 pebbles, especially particle size, impurities, crush load, are introduced in this study.
Development of surface modification of beryllide pebbles with no-hydrogen generation reaction with steam

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Hydrogen generation via an oxidation reaction of beryllium as an existing neutron multiplier with steam at high temperatures should be reduced on safety hazard for a fusion reactor. Therefore, advanced neutron multipliers with high stability at high temperatures are desirable for the fusion reactor in which water coolant is extensively used. Beryllium intermetallic compounds (beryllides) are one of promising materials. Fabrication methods of beryllides pebbles have been developed by combining a plasma sintering synthesis method and a rotating electrode granulation method. In the case of Be12Ti, an annealing treatment is necessary to homogenize the pebbles to a single Be12Ti phase after granulation, because the composition changes via a peritectic reaction caused by remelting under granulation. Homogenized Be12Ti pebbles found out to have higher reactivity than as received pebbles because the homogenization treatment caused to an increased specific surface area of the pebbles. To prevent the increased surface area and lower the reactivity, accordingly, prototypic pebbles with Be12V that have no peritectic reaction during granulation process were successfully fabricated without homogenization. Then, the Be12V prototypic pebbles indicated to have a good oxidation resistance. As a result of the reactivity with steam, it was clarified that BeO layer on the surface of Be12V pebbles acts as a protective barrier against hydrogen generation reaction. As the next stage, to investigate the effect of BeO layer as surface modification of beryllide on the reactivity, Be12V pebbles annealed in oxygen atmosphere were prepared as specimen. Using those surface-modified pebbles, hydrogen generation reaction experiments at 1273 K were repeated three times. Hydrogen generation rate of each experiments reduced to almost the same level as the background. From these results, surface-modified Be12V pebbles with no-hydrogen generation were successfully fabricated and developed.
Li mass loss from Li2TiO3 fabricated by the emulsion method under optimized condition

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Understanding of Li evaporation property is important because Li mass transfer decreases tritium breeding ratio and influences tritium behavior. In JAEA, the development of Li2TiO3 with excess Li has been performed as an advanced tritium breeder. The present authors revealed in previous works that a layer existing on the pebble surface includes Li2CO3 and it contributes Li mass loss. Recently, Li2TiO3 pebbles without the surface layer were successfully fabricated by the emulsion method under optimized sintering condition in JAEA. In this work, Li mass loss property for newly fabricated pebbles was investigated. Li2TiO3 pebbles with/without the surface layer are referred to as old (Li/Ti=2.10) / improved pebbles (Li/Ti=2.10). Sample pebbles were packed in a quartz tube and heated to 900 °C in 50 Pa or 200Pa H2O/Ar. Li mass loss was estimated from the weight change of pebbles before and after the experiment taking account of the water vapor release. SEM observation was carried out before and after the experiment. In the additional experiment, CO2 release was measured by a gas chromatograph. Li mass loss from improved pebbles was 1.0 wt% in 50 Pa H2O/Ar and 1.5 wt% in 200Pa H2O/Ar. Li mass loss from old pebbles was 1.5 wt% in 50 Pa H2O/Ar and 2.7 wt% in 200Pa H2O/Ar. It was found that Li mass loss from improved pebbles was smaller than that from old one. Although CO2 release was observed from old pebbles over 600 °C by the decomposition on Li2CO3, CO2 release was not detected from improved pebbles. Although the melting layer of Li2CO3 was observed on old pebbles after heating over 700°C, Structure changes of improved pebbles were not observed by SEM observation. It can be said that Li mass loss from improved pebbles is progressed without the contribution of Li2CO3.
Lithium metatitanate (Li$_2$TiO$_3$) is one of the candidate materials among the solid tritium breeders proposed because of its good tritium release property and high chemical stability [1]. Lithium metatitanate with excess Li (Li$_{2+x}$TiO$_{3+y}$) has been recognized as an another prominent candidate material owing to its higher Li density [2]. Demonstration power plant (DEMO) reactors require tritium breeders with high mechanical stability under operating conditions. However, their compression fracture strengths are feared to be reduced by Li vaporization during operation at high temperature. In this work, the compression fracture strength changes of the pebbles of Li$_2$TiO$_3$ and Li$_{2+x}$TiO$_{3+y}$ under operating conditions were investigated. The pebbles were thermally annealed in a gas flow of 1%H$_2$/He at 900°C for various time, and the compression fracture strength was measured by using an universal testing machine at room temperature. Mass decrease by Li vaporization was measured by gravimetric method, and crystal-phase change was analyzed by XRD. When the sintering density was below 78%, the compression fracture strength and apparent Young’s Modulus were significantly small. Therefore, the pebbles with sufficiently large and constant sintering density (83−85%) were used in this study. [1] N. Roux, J. Avon, A. Floreancig, J. Mougin, B. Ravel, J. Nucl. Mater., 223-237 (2) (1996) 1431-1435. [2] T. Hoshino, M. Yasumoto, K. Tsuchiya, K. Hayashi, H. Nishimura, A. Suzuki, T. Terai, Fusion Eng. Des., 82 (2007) 2269-2273.
Li vaporization property of candidate materials for tritium breeder with high Li density

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Lithium metatitanate (Li2TiO3) is one of the candidate materials for solid tritium breeder proposed because of its good tritium release property and high chemical stability [1], and Lithium metatitanate with excess Li (Li2+xTiO3+y) has been recognized as a prominent candidate material owing to its higher Li density [2]. However, demonstration power plant (DEMO) reactors require tritium breeders with higher lithium density. From the view point of Li density, Li8ZrO6 and Li5AlO4 are promising candidates for the breeder. However, it is concerned that the high lithium densities of these materials cannot be maintained, because they have high Li vapor pressure. Ogawa et al reported that stable phases with low Li vapor pressure surrounding the unstable phase prevents the Li vaporization from the breeder [3]. In this work, Li vaporization ratios from sintered compacts of the single phases of these candidate materials and the complexes of the high-Li-density phase and stable phase (Li2TiO3) were measured. Mass decrease by Li vaporization was measured by gravimetric method, and crystal-phase change was analyzed by XRD. From some sintered compacts, the ratio of mass decreases by the Li vaporization became small after significant decreases confirmed in an initial 30 h. These results suggest the possibility of a new advanced tritium breeder with high Li density. [1] N. Roux, J. Avon, A. Floreancig, J. Mougin, B. Ravel, J. Nucl. Mater., 223-237 (2) (1996) 1431-1435. [2] T. Hoshino, M. Yasumoto, K. Tsuchiya, K. Hayashi, H. Nishimura, A. Suzuki, T. Terai, Fusion Eng. Des., 82 (2007) 2269-2273. [3] S. Ogawa, Y. Masuko, H. Kato, H. Yuyama, Y. Sakai, E. Niwa, T. Hashimoto, K. Mukai, T. Hoshino, K. Sasaki, Fusion Eng. Des., 98-99 (2015) 1859-1863.
P3.170

Characterisation and radiolysis of modified lithium orthosilicate pebbles with noble metal impurities

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Modified lithium orthosilicate pebbles with additions of titanium dioxide are suggested as an alternative tritium breeding ceramic for the Helium Cooled Pebble Bed (HCPB) Test Blanket Module (TBM). The tritium breeding ceramic in the HCPB TBM will be under the action of harsh operation conditions. Radiolysis can take place as a result, and unstable radiation-induced defects (RD) and radiolysis products (RP) can form. The formed RD and RP can interact with the generated tritium and may disturb tritium diffusion and hinder its release. In this research, the influence of the noble metal impurities (platinum, gold and rhodium which are introduced during the production process) on the radiolysis of the breeder pebbles was analysed. High energy accelerated electrons were used instead of neutron irradiation to avoid radiolysis effects in order to introduce radiolysis in the HCPB TBM will be under the action of harsh operation conditions. Radiolysis can take place as a result, and unstable radiation-induced defects (RD) and radiolysis products (RP) can form. The formed RD and RP can interact with the generated tritium and may disturb tritium diffusion and hinder its release. In this research, the influence of the noble metal impurities (platinum, gold and rhodium which are introduced during the production process) on the radiolysis of the breeder pebbles was analysed. High energy accelerated electrons were used instead of neutron irradiation to introduce radiolysis effects in order to avoid nuclear reactions and thereby the formation of radioactive isotopes. The samples were irradiated with accelerated electrons (E=5 MeV, D=12 MGy, T=300-345 K, dry argon). The irradiation parameters were selected, to accumulate mainly primary and secondary RD. The formation and accumulation of RD were subsequently analysed by electron spin resonance (ESR) spectroscopy. Also, the chemical composition, noble metal concentration and surface microstructure of the pebbles were studied. Using ESR spectroscopy, the formation and accumulation of several paramagnetic RD were detected, such as E' centres, HC2 centres etc. It was determined that the trace-impurities of the noble metals, with a sum content of up to 300 ppm, do not significantly influence the formation and accumulation of RD in the modified lithium orthosilicate pebbles. Acknowledgment: This research of the Baltic-German University Liaison Office was supported by the German Academic Exchange Service (DAAD) with funds from the Foreign Office of the Federal Republic Germany. The views and opinions expressed herein do not reflect those of the Baltic-German University Liaison Office.
Discrete element modelling of ellipsoidal particles for fusion applications

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Five ITER project members are actively involved in the fabrication of tritium breeding ceramics pebbles. Different fabrication processes developed by these members strongly influence the characteristics of pebbles produced. One of the main characteristics is the sphericity of pebbles. The spherical shape is the one desired; however the manufacture of perfect round particles is not simple. For these pebbles, sphericity is usually different from unity, sometimes it is quite high (i.e. needle type particles) but sometimes the shape is more close to an oblate spheroid. A Discrete Element Method (DEM) code has been previously developed in the Institute of Applied Materials (IAM) in order to simulate the mechanical behavior of a fusion pebble bed, where pebbles were simulated by perfectly spherical particles. In order to simulate the mechanical behavior of ellipsoidal particles an extension of the previous DEM code was necessary. The method taken into account to represent non spherical particles was the multi-sphere (MS) approach. It is a multi-particles approach based on the union of several spheres to obtain the required shape. The MS method leaves the possibility to continue to use the same algorithms developed for spherical particles. Recently, further improvements have been brought to guarantee the versatility of the code. The code was generalized to allow the user to choose the number and the radius of spherical particles that compose the ellipsoids. Sensitivity studies, mainly related to the variation of the aspect ratio and the radius of spheres, were performed in order to guarantee the applicability of the code for different kinds of pebble assemblies. The aim in the future is to extend the algorithm to even more generalized particle shapes.
Displacement damage effect on the thermal stability of deuterium in SiC

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SiC is a primary candidate for flow channel inserts in blankets due to their excellent thermo-mechanical properties. During reactor operation SiC will be exposed to tritium in a hostile radiation environment. Absorption, diffusion, and desorption will occur, and are expected to depend on the neutron and ionizing radiation conditions. We present work to assess the effect of displacement damage on the thermal stability of deuterium in reaction bonded (RB) SiC in the temperature range 400 to 1000 °C. The aim is to study the thermal stability of deuterium as a function of displacement damage induced with 50 keV Ne⁺⁺, either before or after deuterium implantation. Three experiments were carried out: first 10 keV D₂⁺⁺ ion implantation at 450 °C in SiC samples for reference; second, pre-damage of SiC samples with 50 keV Ne⁺⁺, followed by deuterium implantation; and third in the reverse order for further SiC samples (deuterium implantation, followed by damage with Ne⁺⁺). After these treatments a sample of each method was examined using SIMS. A further three samples were heated up to 1000 °C at a rate of 0.16 °C/s, in order to obtain the deuterium thermally stimulated desorption (TSD) spectra. Following TSD measurements, these samples were examined using SIMS. TSD and SIMS analysis of reference samples show the deuterium retention in SiC is higher for the RB SiC pre-damaged with Ne⁺⁺ and also that the thermal stability of the deuterium is enhanced by traps induced by the Ne⁺⁺ damage. In contrast for the post-damaged samples, no difference was found compared with reference samples. This work shows that the displacement damage produced into SiC generates thermally stable traps for hydrogen isotopes. Following the initial trapping process, additional damage does not play an important role on the overall retention.
Radiation exposure effect on deuterium retention in SiC

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Silicon carbide (SiC) is considered to be used for blanket modules for high temperature gas–cooling system in D-T fusion reactors, as SiC/SiC composites. During D-T fusion operation, SiC will be exposed to heavy radiation conditions by neutron and/or gamma-ray. These radiation induces the formation of various damages by a collision process and an electron excitation process, leading to the retention enhancement of hydrogen isotopes including tritium. In this study, 6.4 MeV Fe3+ and/or gamma ray irradiations were performed for SiC and thereafter D$^+$ implantation or D2 gas exposure experiment was done. Their D retention and chemical behaviors were evaluated by thermal desorption spectroscopy (TDS) and X-ray photoelectron spectroscopy (XPS). Disk type beta-SiC was used as a sample. After annealing at 1173 K, 6.4 MeV Fe3+ implantation with a damage concentration of 0.2 dpa was performed at DuET tandem accelerator at Kyoto University. Cobalt-60 gamma ray irradiation was performed up to the dose of 400 kGy. Thereafter, 1 keV D$^+$ implantation with a fluence of 1.0×1022 D$^+$ m$^{-2}$ was performed at room temperature. In addition, D2 gas exposure experiment was also performed at 100 kPa for 20 hours. The TDS measurement showed that D2 desorption was consisted of two stages located at 890 K and 1080 K, attributing to be the desorption of D bound to Si as Si-D bond and that bound to C as C-D bond, respectively. For radiation exposed SiC, both of D retentions as both stages 1 and 2 were increased as D$^+$ fluence increased, which was quite different from that for undamaged SiC, suggesting that the formation of dangling bonds enhanced D trapping efficiency. The detail D trapping behavior and change of chemical state for SiC was discussed in the presentation.
Study on the corrosion behavior of CVD and CVI SiC materials with liquid Pb–Li under rotating flow

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Lead–lithium (Pb–Li) alloy are considered as a coolant and a tritium breeder for fusion reactor blanket systems. One of the critical requirements for the realization of this systems is the compatibility of silicon carbide (SiC) and its composites as structural and/or functional materials. The authors investigated that inclusions, possibly Li–oxides in Pb–Li may have certain impacts on compatibility of the materials through reactions with oxides in/on materials even if the oxygen level in Pb–Li is low. This study aims to clarify the degradation behavior of CVI–SiC with liquid Pb–Li using rotating disk systems under rotating flow condition at 700°C up to 3,000h. Key-words: silicon carbide (SiC), chemical vapor deposition (CVD), chemical vapor infiltration (CVI), liquid breeding material, Pb–Li, rotating flow condition
Radiation induced deuterium absorption dependence on temperature, dose rate, and gas pressure for SiC

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During ITER and DEMO reactor operation the proposed Li-Pb blanket flow channel inserts made of SiC ceramic material will be exposed to both radiation and tritium. Absorption, diffusion, and desorption of tritium is expected to occur and these processes will strongly depend on the irradiation conditions, neutron flux, and purely ionizing radiation. Previous results have shown that marked deuterium absorption, associated with the formation of silicon deuterium bonding, occurs for SiC materials when both deuterium and sample are subjected to a radiation field, and that this radiation enhanced absorption strongly depends on both the displacement damage and the ionizing radiation field. In the work to be presented the roles played by irradiation temperature, dose rate, dose, and deuterium gas pressure have been addressed for reaction bonded SiC. The samples have been irradiated making use of a special chamber with a 50 mm thick aluminium window mounted in the beam line of a Van de Graaff accelerator. The chamber, filled with deuterium gas at different pressures, contains a sample holder with an oven allowing one to heat the samples from room temperature up to 800 C. Both the deuterium gas and samples were irradiated with 1.8 MeV electrons at different dose rates, doses, gas pressures, and sample temperatures. Following irradiation each sample was remounted in another system which permitted one to linearly heat the sample and measure the release rate of any radiation induced absorbed deuterium as a function of temperature. The results show that radiation induced deuterium absorption depends linearly on total ionizing dose and deuterium gas pressure, but not on dose rate. Behaviour with irradiation temperature is more complex, and clear changes in the deuterium thermal desorption are observed to occur depending on irradiation temperature. SIMS results for high temperature loaded RB SiC are consistent with radiation enhanced diffusion.
P3.176

Benchmark experiment on copper with graphite by using DT neutrons at JAEA/FNS

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In our previous copper benchmark experiment we had pointed out that the elastic scattering and capture reaction data of the copper had included some problems in the resonance region, which had caused a large underestimation of reaction rates of non-threshold reactions. In order to corroborate this issue, we carried out a new benchmark experiment on copper in the neutron field with more low energy neutrons. The experimental assembly consisted of quasi-cylindrical copper (63.0 cm diameter, 60.8 cm thick) with graphite blocks (63.0 cm diameter, 10.1 cm thick) in the front part of the copper. Also, the side and rear parts of the assembly were covered with Li$_2$O blocks (5.1 cm, 15.3 cm thick) to absorb background neutrons. We measured reaction rates, $^{93}$Nb$(n,2n)^{92}$mNb, $^{27}$Al$(n,\alpha)^{24}$Na, $^{115}$In$(n,n')^{115m}$In, $^{197}$Au$(n,\gamma)^{198}$Au and $^{186}$W$(n,\gamma)^{187}$W, using the activation foils. We analyzed the experiment with MCNP-5.140 and the latest nuclear data libraries, ENDF/B-VII.1, JEFF-3.2 and JENDL-4.0. As a result, the calculated reaction rates related to low energy neutrons, $^{197}$Au$(n,\gamma)^{198}$Au and $^{186}$W$(n,\gamma)^{187}$W, excessively underestimated the measured ones as in the previous benchmark experiment. We also tested the nuclear data of copper modified in the previous study, where the elastic scattering cross section of copper was increased by 10 % and the capture reaction of copper was decreased by 10 % from 100 eV to 300 keV. Then the calculated reaction rates with the modified copper nuclear data reproduced the measured ones well. It was revealed that the modification of the specific cross sections had been sufficient in the neutron field with more low energy neutrons. The cross section data of the elastic scattering and (or) capture reaction of copper should be reevaluated.
Benchmark experiment on molybdenum with graphite by using DT neutrons at JAEA/FNS

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In our previous benchmark experiment on molybdenum at JAEA/FNS, we found problems of the (n,2n) and (n,γ) cross sections in Mo of JENDL-4.0. However, the Mo data only above a few hundred eV were investigated, because there were few neutrons with lower energy in the Mo assembly in the previous experiment. We perform a new benchmark experiment on Mo in order to validate the Mo data in the lower energy region. A rectangular Mo assembly, the size of which is 253 mm × 253 mm × 354 mm, is covered overall with 152 mm thick graphite blocks. Furthermore, the assembly is covered with 51, 101 and 101 mm thick Li2O blocks around the front, side and back surfaces, respectively. The graphite blocks produce neutrons with the lower energy and the Li2O blocks eliminate background neutrons at the measurement points inside the assembly. The assembly is placed at a distance of 103 mm from a DT neutron source. Several dosimetry reaction rates and fission rates are measured in the assembly and compared with the calculated values with the Monte-Carlo transport code MCNP5-1.40 and the recent nuclear data libraries, ENDF/B-VII.1, JEFF-3.2 and JENDL-4.0. It is suggested that the (n,γ) cross section of $^{95}$Mo is underestimated in the tail region below the large resonance at 45 eV in these nuclear data libraries from the comparison of the reaction rate for the $^{186}$W(n,γ)$^{187}$W reaction which have a large resonance at 19 eV. Reasons of the underestimation are discussed in detail.
Materials from the group of layered Mn+1AXn phases are new type of nanolaminates which can be used in many technical applications, especially as viable candidates for high-radiation structural application in fusion technology. It has been proposed that the novel physical properties of MAX phases arise from their atomic structure, combining “ceramic” MX6 octahedra layers with a single intercalated “metallic” layer, where M is an early transition metal, A is a group III or IV element and X either C or N. The goal of this study is to investigate the microstructure and mechanical properties of the Ti2SnC with layered crystal structure as one of the most fascinating “211” member of MAX phases. We report on the synthesis of Ti2SnC MAX phase with A being low melting metal announced to exhibit superior machinability and excellent thermal shock resistance. Ti2SnC MAX phase was synthesized by powdered Ti, Sn and TiC in a stoichiometric ratio, compressed into a tablet and annealed in a quartz tube under vacuum at 1200°C. SEM was provided in order to obtain information of Ti2SnC surface morphology. The microstructure was analyzed by HRTEM and by EDS microanalysis. The nanoindentation was performed in order to mechanically quantify the grain behavior. Young’s moduli were calculated from unloading parts of the penetration curves. The electron microscopy and the nanoindentation analysis of Ti2SnC sample confirmed the effect of porosity on different scales and two distinct grain types as well. The cyclic indentation confirmed that the material is very compact and the occurred inelastic hysteresis leads to influence of the elastic and hardness parameters. We observed dense Ti2SnC material with only minor fractions of unreacted TiC0.5 allowing its application at extreme and nuclear conditions.
Yttrium oxide coatings as tritium permeation barriers

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In fusion power plants a tritium permeation barrier is required in order to prevent the loss of the fuel inventory. Moreover, the tritium permeation barrier is necessary to avoid that the radioactive tritium accumulates in the first wall, the cooling system, and other parts of the power plant. Oxide thin films, e.g. Er2O3 and Y2O3, are promising candidates as tritium permeation barrier layers. With regard to the application, this is especially true for Y2O3, due to its favorably low activation behavior, compared to the other candidates. Y2O3 thin films are deposited on the reduced activation steel Eurofer97 by means of magnetron sputtering. The thin films are annealed at 550°C to achieve a stable and homogeneous cubic phase of the Y2O3 system. X-ray diffraction analysis proves that the final phase of the thin films is actually cubic. To be able to quantify the permeation reduction factor of the Y2O3 thin films a new gas-driven deuterium permeation measurement setup has been constructed. Comparing the permeation flux through a bare substrate and a coated Eurofer97 substrate, the permeation reduction factor can be determined. The measurement result suggests that the permeation reduction factor is in the range of one hundred.
On the phase transitions of Y2O3 in ultrafine grained W-Y2O3 composite

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Pure tungsten is considered as the most suitable plasma facing material for the reactor first wall. However, number of studies points out serious drawbacks related to tungsten mechanical properties that negatively affect lifetime of first wall components. Serious risk for the divertor comes from abnormal events, such as disruptions, vertical displacement events (VDEs) and edge localized modes (ELMs). The transient loading can deposit large energies on the divertor surface (power density $\approx GW/m^2-2$) within a period of few ms. Thus, most serious concerns about tungsten are related to the thermally induced grain growth. The abovementioned disadvantages lead to current efforts to develop tungsten with improved properties. Among the most studied alternative candidates from the family of tungsten with oxide dispersion is W-Y2O3. Various concentrations starting with values as low as 0.1% up to 5 wt% of Y2O3 and related mechanical properties are studied. Nevertheless, it seems no attention is given to the characterization of the Y2O3 phase in the prepared composites despite the significant density and microstructural difference of the different Y2O3 forms. Thus, the phase transition is accompanied by a volume change which might significantly alter properties of the prepared W-Y2O3 composite. Y2O3 is a polymorph that can occur in three basic forms: cubic, monoclinic and hexagonal. Presented study brings description and identification of the condition for phase transitions and structure changes of Y2O3 in W-Y2O3 composite from the production to the application in the future fusion reactor. The phases were characterized by XRD, HRTEM and Raman spectroscopy. The causes of the phase transition are discussed and the range of the phase stability for the different Y2O3 forms is clarified.
Impact of pretreatment conditions on defect formation during the fabrication of Al-based corrosion barriers

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Different breeding blanket designs for a future fusion power plant (DEMO) consider Eurofer steel as a main structural material. Nevertheless, RAFM steels suffer from severe corrosion attack in Pb-15.7Li, which acts as breeding material in the liquid breeder blanket designs, e.g. HCLL, WCLL and DCLL. The resulting corrosion products may cause safety risks e.g. concerning tube plugging due to possible precipitation of dissolved steel components. Two electroplating processes were developed in the past to produce protective Al-based coatings on RAFM steels. These coatings proved that they could protect the underlying RAFM steel from corrosion in flowing Pb-15.7Li at fusion relevant conditions. In this context electroplating processes i.e. ECA and ECX, showed favorable characteristics compared to hot-dip aluminization. Beyond an adapted heat treatment, both electrochemical processes need reliable pretreatment processes of the RAFM substrates prior to the Al-deposition, to prevent coating defects such as insufficient covering and weak adhesion. These coating failures increase the risk of defects in the corrosion barriers after the heat treatment and therefore may promote early coating breakdowns in flowing Pb-15.7Li environment. However the influence of e.g. the Eurofer substrate condition prior to the pretreatment and the duration of the pretreatment, on the coating appearance and homogeneity were not examined in detail until now. This study examined these influences on defect formation by electrochemical measurements and SEM/BSE examinations. Besides storage time between mechanical preparation of the samples and electrodeposition, the duration of the anodic pretreatment of Eurofer samples was varied prior to the Al-plating by ECX process. It was shown that the degree of covering of aluminum on RAFM steel substrates depended on both parameters. From these findings optimized pretreatment parameters were derived that increase the reliability of the whole Al-based barrier fabrication process to achieve improved Fe-Al coatings with smooth and dense surfaces with uniform properties.
Fatigue life of tungsten materials strengthened by various methods

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Tungsten (W) is a primary candidate for fusion reactor divertor because of its high melting point, thermal conductivity and sputtering resistance. To improve its structural reliability, improvement of mechanical properties and suppression of recrystallization of the W materials are necessary. It is well known that the grain refining, work hardening, solid solution strengthening, and dispersion strengthening are the methods for them. Recently, we have developed various W materials, which were strengthened by a potassium (K) doping for dispersion strengthening and a rhenium (Re) addition for solid solution strengthening. These materials showed advantageous tensile properties and recrystallization temperature in comparison with the conventional pure W plate. Though one of the most important mechanical properties for divertor material is fatigue, limited experimental data on the effect of these strengthening treatments are reported. The objective of this study is to investigate the low cycle fatigue life of these W materials and to clarify the effect of the K-doping and Re addition. Materials evaluated were pure W plate, K-doped W plate, K-doped W-3\%Re plate, and K-doped W rod. Low cycle fatigue tests were carried out at 500°C in vacuum under axial stroke control. A completely reversed push-pull condition was applied. The total strain range was $0.7\%[\pm 1.3\%]$, which was estimated from the stroke range value and mechanical compliance of the testing machine. The effect of the K-doping on the fatigue life was observed in the hot-rolled plate material. The effect of the K-doping and Re addition will be discussed.
Diffusion-controlled F center thermal annealing in neutron, electron and heavy-ion irradiated insulators

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The radiation-resistant insulators (MgO, Al2O3, MgAl2O4, BeO etc) are important key materials for fusion reactors. It is very important to predict/simulate not only the kinetics of diffusion-controlled defect accumulation under neutron irradiation, but also a long-time defect structure evolution including thermal defect annealing. Here we developed and applied the advanced theoretical approach based on the formalism of the correlation functions suited much better for the study of defect kinetics and aggregation than generally accepted rate equations. On the basis of our calculations, we estimated the migration energy of the F centers and interstitial oxygen defects Oi, their interaction energies and metal colloid size upon annealing. We simulated the F-type center annealing after electron, heavy ions or neutron irradiation as a bimolecular process with equal concentrations of the complementary F and Oi defects. It is controlled by the interstitial oxygen ion mobility, which is much higher than that of the F centers. The F center annealing begins at temperatures 500-700 K, when both F and F+ centers are practically immobile, due to the recombination with mobile Oi defects. It is demonstrated how the shape of the F-annealing curves is determined by two control parameters: Ea and effective pre-exponential factor and strongly depends on irradiation conditions. The appropriate migration energies were obtained from available in literature annealing kinetics for electron, neutron and ion-irradiated oxide crystals (MgO, Al2O3, MgAl2O4, BeO, ZnO, PLZT etc). The results obtained are also compared with recent ab initio calculations of interstitial oxygen migration (MgO and Al2O4).
Formation of Cr2O3 layers on coolant duct materials for suppression of hydrogen permeation

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In our previous study, a Cr2O3 layer was formed on a reduced activation ferritic/martenitic (RAFM) steel substrates by heat treatment under a reduced atmosphere and it could suppress hydrogen permeation by \~2 orders at 550-650 °C. Since the Cr2O3 layer was stable at high temperatures in air, it was also a preferable underlayer for multi-layer ceramic coating with the metal organic decomposition method. In the present study, formation of Cr2O3 layers on candidate duct materials for liquid blanket systems, i.e. HASTELLOY X, INCONEL 600 and SUS316L, are examined. The substrates were heat treated in a vacuum of 5 Pa at 700 °C for 1 hour. During the heating up and cooling down processes, the pressure was kept at <5x10^{-3}-3 Pa to suppress formation of Fe2O3 which degrades permeation barrier performances. On the HASTELLOY substrate, a \~150 nm thick Cr2O3 layer was formed. The thickness and the blue-colored surface were similar to those previously obtained on RAFM steel substrates. On the INCONEL substrate, a thinner Cr2O3 layer of \~50 nm was obtained. These Cr2O3 layers on the Ni-based alloys are expected to suppress hydrogen permeation and to be stable underlayers in multi-layer ceramic coating fabrication. However, a thick Fe2O3 layer of >200 nm was formed on a SUS316L substrate. Although the composition of the surface layer was improved to 60 at% Fe2O3 and 40 at% Cr2O3 by treatment at 800 °C, further improvement of the condition will be required to suppress the Fe2O3 production. Characterization of the Cr2O3 layers including hydrogen permeation barrier performances is being performed at present.
Deuterium permeation behavior of tritium permeation barrier coating containing carbide nanoparticles

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Tritium permeation barrier (TPB) has been investigated for the establishment of an efficient fuel cycle and radiological safety in fusion power plants. One of critical issues for TPB is degradation caused by introduction of cracks and pores. Even if a microscopic crack is introduced, tritium permeation is drastically increased. The development of self-healing coating is one of techniques for solving this issue. In this study, the self-healing material was selected through oxidation experiments. Fabrication and deuterium permeation measurements of TPB coatings containing the self-healing material were carried out to examine their basic properties as TPBs. SiC and Cr3C2 powders were heated at 773-973 K for 1-10 h in air followed by crystal structure analysis using X-ray diffraction. Since SiC was not oxidized at up to 973 K while Cr3C2 was oxidized at 873 K, Cr3C2 was selected as the self-healing material. Subsequently, yttria coatings with Cr3C2 nanoparticles have been fabricated by metal organic decomposition. Deuterium permeation experiments were performed at 673-973 K using a gas-driven permeation system. Two types of coated samples were fabricated: Cr3C2 nanoparticles were added in a coating precursor in the first coating process (Sample 1), and the yttria coating without the nanoparticles was fabricated followed by the addition of nanoparticles in the second coating process (Sample 2). A decrease of deuterium permeability caused by crystallization was observed for both samples at 623-973 K. After the crystallization, Sample 1 showed two orders of magnitude lower deuterium permeation than that for uncoated substrate, indicating a lower surface coverage. On the other hand, Sample 2 showed three orders of magnitude lower deuterium permeation than that for uncoated substrate. Pretreatment of substrate is effective for the fabrication of the nanoparticle-containing coatings to ensure surface coverage.
Deuterium permeation and retention behaviors in erbium oxide-iron multilayer coatings

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To establish liquid lithium-lead blanket concepts, the development of a functional coating as a tritium permeation barrier with corrosion resistance is required. In our previous study, erbium oxide (erbia)-iron two-layer coatings showed a better compatibility than erbia single-layer coatings with keeping a high permeation reduction factor (PRF). In this study, hydrogen isotope migration behaviors in ceramics-metal multilayer coatings have been elucidated for a further improvement of fueling system and radiological safety. First, erbia coatings (thickness: 1–3 μm) were fabricated by filtered vacuum arc deposition (VAD) on reduced activation ferritic/martensitic steel F82H substrates. Second, iron layer was fabricated by radio-frequency magnetron sputtering (1 μm) or covered with an iron foil (10 μm) on the erbia coating. An erbia-iron-erbia three-layer coating was also fabricated by the VAD on the erbia-iron coating. Subsequently, deuterium permeation experiments were carried out in the temperature range of 773–973 K. After the permeation tests, deuterium was introduced into the samples at 873 K with 80 kPa deuterium. Depth profiles of deuterium concentration in the coatings were evaluated by the D(3He, p)4He nuclear reaction. The erbia-iron coated samples with different iron layer thickness showed no significant difference in deuterium permeability, indicating that deuterium permeation in the erbia-iron was controlled by diffusion in the erbia layer. The erbia-iron-erbia coating showed a PRF of up to 10^4 due to a contribution of two diffusion barriers of inner and outer erbia layers. However, the PRF was less than that of the erbia sample coated on both sides of the substrate, possibly derived from a different recombination process on the back surface. Moreover, the erbia-iron-erbia coating had three times higher D concentration than the erbia-Fe coating (1.9 × 10^{15} atom cm^{-2}), which suggests tritium inventory in the multi-layer coatings should be taken into account.
Survey of oxide candidate for advanced 9%, 14% and 17%Cr ODS steels for fusion applications

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The structural components used for construction of future generation of fission reactors and fusion reactors will undergo demanding service conditions as high neutron doses, high temperature and extremely corrosive environment. The nano-structured oxide dispersion steels (ODS) containing small amounts of homogeneously dispersed nano-size yttria particles were developed as structural material for fast breeder reactors. Three classes of prospective structural ODS steels can be identified: ferritic-martensitic 9%Cr steel and ferritic 14%Cr and 17%Cr steels. The aim of this work was to find new candidate low-activation ODS steels strengthened by alternative nano-sized oxides. The 9Cr-1W, 14Cr-2W and 17Cr-1Mo ODS steels containing oxides based on Y, Al, Zr, Ti and Ca were prepared by mechanical alloying from atomic powders. The oxide dispersion was created alternatively by controlled oxidation powder during mechanical alloying process. Fully dense steels were obtained after densification of prepared powders by spark-plasma-sintering process. Structural and mechanical properties of the new candidate ODS steels were compared to conventional oxide-free steels and ODS steels strengthened by direct adding of relevant oxides. The new procedure for preparation of nano-structured ODS steels was developed based on internal oxidation of oxide-trapping atoms. The oxides were created from oxygen and Y, Al, Zr, Ti and Ca atoms added to the alloy during mechanical alloying of the powder. Oxide distribution increases tensile yield strength and ultimate tensile strength both at room and elevated temperature without distinct deterioration of plasticity.
Fracture toughness of the new generation ODS steels with different oxide compositions

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ODS steels are candidates for the structural material in the future fusion power plants. Their main advantage is high strength and creep resistance at high temperatures. Such high performance is achieved by the presence of the oxide particles in the microstructure. Nowadays, the best ODS steels contain particles of Y2O3 which are stable at high temperatures. However, yttrium is expensive and its reserves are limited. Therefore, the new generation of the ODS steels has been developed and tested in our institute. Steels with 9 and 14% of chromium and different oxide compositions (Zr, Ti, Al) are tested. The main objective is to further improve the steels properties and test other elements which can replace yttrium. Fracture toughness is an important variable for construction materials. The current study evaluates fracture toughness for these new steels in the wide temperature range (-80°C - 600°C). The results are compared with up to date most advanced ODS steels 12YWT and 14YWT. The newly developed steels show higher fracture toughness and lower brittle-ductile transition temperature. Such results are promising in the future development of structural materials for high temperature applications and particularly fusion power plants.
Tungsten/Steel Composites for the application in Functionally Graded Interlayers at the armor-substrate transition zone of the first wall

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Future fusion reactors may exhibit first walls composed of a tungsten (W) armor, that is attached to a subjacent stainless steel (SS) structure. Joining these materials for the application at hand is challenging because the pulsed operation of TOKAMAK reactors induces thermo-mechanical stresses and strains at the W/SS interface due to differing materials properties. These cyclic loads will degrade the W/SS joint, eventually resulting in delamination of the armor, in which the number of endured cycles strongly depends on the type and quality of the W/SS bond. Current approaches focus on implementing a functionally graded (FG) interlayer between the W and the SS part. The FG composite exhibits a varying fractional W/SS composition along the interlayer height, thus re-distributing and reducing macroscopic loads, as compared to a direct W/SS joint. In order to characterize FG interlayers, this contribution addresses the microstructure as well as thermal and mechanical properties of several W/SS compositions manufactured via Atmospheric Plasma Spraying (APS) and Electro Discharge Sintering (EDS). In this context the residual porosity, the formation of phases, thermal conductivities as well as thermal expansions, and the fracture toughness at temperatures ranging from room temperature to 300°C were investigated.
Fusion reactor is one of new type reactors being developed, and it is cleaner and more efficient than the fission reactor. Each SSCs (Structures, Systems, Components) has different safety importance to fusion reactors. So it is necessary to classify the SSCs of fusion reactors. And the safety classification of SSCs for fusion reactor is the important basis of reactor design and construction. Now the safety classification of SSCs for reactors is too conservative and it cannot identify all the important SSCs for fission and fusion reactors. Probabilistic safety assessment (PSA, the frequently-used PSA codes i.e. RiskA) is one of the important tools to evaluate whether the reactor is safety or not. The importance and sensitivity of SSCs can be calculated by PSA. And the PSA method can be used to identify whether the SSCs is important or not according to the numerical value of importance and sensitivity. A new safety classification approach for fusion reactors is proposed in this paper, in which PSA method is incorporated. The PSA should be performed and the importance and sensitivity of the SSCs can be obtained according to the results of PSA. The value of importance and sensitivity reflects the importance degree for SSCs from different angles, and a sequence of the SSCs can be performed according to the numerical value of importance and sensitivity. The numerical value will be different from fission reactors to distinguish the safety class SSCs and non-safety class SSCs. The SSCs can be classified according to the sequence of the SSCs and it should be more rational to consider the classification from the engineering experience. Through the above process, the final safety classification is obtained from probabilistic point, and this method is called probabilistic safety classification. This safety classification reflects the importance of SSCs from the probabilistic points.
Research of Failure Data Adjust Method and Application for Tritium Extraction System

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The high reliability and availability of Tritium extraction system (TES) will be needed is necessary for safety operation of circulation and processing of tritium purge gas. Reliability, availability, maintainability, inspectability (RAMI) analysis of the TES should be performed during the design and operation phase. Since there is no TES failure rate data available from fusion operating experiences. Therefore, it is necessary to use the existing component operating experiences to calculate component failure rates of the TES components. In this paper, a combinational method was used to adjust component failure rate of TES, based on k factors method and Bayesian method. The method consists of three procedures: 1) the general reliability data gained from the long-term operation in nuclear plants which was chosen as the basic failure data of components. 2) The calculation of each individual k factor was carried out according to the environments and conditions known to affect component reliability, including the operating temperature, the neutron radiation damage, the flow and flow media, the pressure and the vibration and so on. 3) The hyperparameters of failure distribution were calculated based on a Bayesian approach with a Jeffreys noninformative prior distribution. The failure rate values of components were selected from the RiskA/RiskBase (Database Management System for Reliability Analysis) and FCFR-DB (Fusion Component Failure Rate Database). TES components failure rate were predicted based on the combinational method. And then the TES RAMI analysis was also performed based on these adjusted failure rates. This paper introduced a combinational method, which could be applied to failure rates adjustment of advanced nuclear system components.
A preliminary RAMI (reliability, availability, maintainability and inspectability) assessment for the EAST in-vessel components cooling system based on currently available design is presented. The following sub-systems were considered in the analysis: the EAST PFCs heat-sink cooling system, two water pumps system, cooling loop including cycle feed pipe and cycle return pipe lines, secondary cooling equipment and pressure control system (PCS). EAST in-vessel components cooling system criticality chart was taken into discussion through Failure Mode, Effects & Criticality Analysis (FMECA), the criterion and program based on the ITER organization defined criticality, occurrence and severity rating scale. The reliability block diagrams (RBD) models were implemented taking into account: system reliability-wise configuration, operating schedule currently foreseen on the EAST experiment planning, maintenance schedule and plant evolution schedule as well as failure and corrective maintenance models. A simulation of plant activity was then performed on implemented RBDs to estimate plant availability performance on a mission time of 20 calendar years. The resulting availability performance was finally compared to availability goals previously proposed for DEMO plant by a panel of experts, also compared to the ITER divertor availability goals. The study suggests that inherent availability goals proposed for EAST in-vessel components cooling system are potentially achievable; the most critical failure modes of cooling system were highlighted during EAST operation, at the same time the mitigation actions for major risk were recommended. A sensitivity analysis is also presented to explore results dependency on key estimated parameters and analysis assumptions.
Operational behaviour of a passive auto-catalytic recombiner under low pressure conditions

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In case of a severe accident inside the ITER fusion facility, there exist several scenarios in which hydrogen may be produced and released into the suppression tank. Assuming the accidental ingress of air, the formation of flammable gas mixtures may lead to explosions and severe component failure. One option to mitigate such hypothetical scenarios is the installation of passive auto-catalytic recombiners (PARs), which are presently used as safety devices inside the containments of nuclear fission reactors. PARs convert hydrogen into water vapour by means of passive mechanisms and support the prevention of large accumulations of combustible gases. In cooperation between IRSN (France), Forschungszentrum Jülich (Germany) and RWTH Aachen University (Germany), the operation of PARs inside the suppression tank under accident conditions is investigated. Experimental investigations of PAR operation have been performed with a scaled-down model of a conventional PAR inside the REKO-4 facility (Jülich), a pressure vessel with a volume of 5.3 m³. A first low-pressure test series has been performed with gauge pressures between -0.8 and 0 bar, and hydrogen concentrations of up to 6 vol.%. The test results show a strong dependence of the pressure on the PAR start-up behaviour. The start-up delay is proportional to the pressure level. Furthermore, the recombination rate is significantly reduced with decreasing pressure. The experimental results provide valuable information on PAR operation under challenging conditions for model development and the assessment of PAR performance under accidental conditions.
Reference accident sequences for a demonstration fusion power plant

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Safety studies are performed in the frame of the conceptual design studies for the European DEMO reactor to assess the safety and environmental impact of design options. An exhaustive set of reference accident sequences are defined in order to evaluate plant response in the most challenging events and compliance with safety requirements. The identification of a comprehensive set of accident initiators is the first step for the definition of the reference accident sequences. The Functional Failure Mode and Effect Analysis (FFMEA), based on a top-down approach, is a suitable methodology to define possible accident initiators when insufficient design detail is available to allow for more specific evaluation at component level. The main process, safety and protection functions related to the DEMO reactor are defined through a functional breakdown structure (FBS). Then, an exhaustive set of high level accident initiators is defined referring to loss of functions, rather than to specific failures of systems and components, overcoming the lack of detailed design information. Nonetheless reference to systems or main components is always highlighted, as much as possible, in order to point out causes and safety consequences. Through the FFMEA a complete list of potential accident initiating events (IEs) is provided together with suggestions to improve the overall safety of the machine. From the complete list of IEs, a set of postulated initiating events (PIEs) is selected as the most representative in terms of challenging conditions for the safety of the plant. All the four blanket concepts of the European DEMO reactor are analysed by the FFMEA and a first set of reference accident sequences are selected by a collaboration among European Research Units. The main goal of this paper is to outline the reference accident sequences selected for the four blanket concepts of DEMO.
The Development of an assessment framework to evaluate DEMO plant concept options

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The generation and investigation of alternative design solutions and their benchmarking against criteria that are traceable to high level objectives is a fundamental facet of a holistic systems engineering approach. During the pre-conceptual design phase of DEMO, characterisation studies for multiple plant concepts are being conducted in parallel to explore the design space and evaluate the potential of alternative solutions to areas of plant design where there are known feasibility issues. A systematic assessment of these candidate plant options and selection of the candidate that has the greatest potential to meet requirements for a given risk acceptance, is perhaps the step that will most fundamentally determine the probability of the programme delivering a design that satisfies the requirements overall requirements. The Eurofusion PPPT PMU is therefore developing a Plant Concept Assessment Framework that shall provide a robust and traceable assessment of DEMO plant concepts against assessment criteria. The framework shall incorporate criteria that represent the full breadth of concerns that are of importance to stakeholders, encompassing area such as safety, plant performance, economic factors, technical risk, environmental & sustainability and timescale to deployment. This framework is being developed in collaboration with industry and will take the lead in performing targeted studies to assess technology assumptions that under-pin a given DEMO concept. The methodology seeks to follow established approaches developed for the evaluation of innovative nuclear systems in fission, but will be tailored to meet the specific requirements of DEMO concept selection. This assessment framework will perform an important function as DEMO transitions from pre-conceptual to conceptual phase, through facilitating plant concept selection and key system and technology decisions. The paper provides an overview of the framework methodology, the selection criteria and the plan for implementation.
Improved fusion reactor designs as per integration of advanced systems code modules

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A fusion power plant is characterized by many subsystems operating under extreme thermal and nuclear conditions, thus compelling to be designed according to physics and engineering constraints. For such an operation, dedicated tools called systems codes are currently used. At Karlsruhe Institute of Technology (KIT), a dedicated modelling campaign has been recently launched aiming to study the technology aspects of the key reactor’s components, such as breeding blanket and magnets system. Mostly due to computing time constraints, modern systems codes are based on rather simplified mathematical models. The idea behind these activities is to enhance, compared to existing systems codes, the level of details for the implemented models, for instance in terms of geometrical characterization and modelling sophistication of the simulated reactor’s elements. This approach is expected to catch more accurately some of the key issues affecting the power plant design, though avoiding massive and time consuming full scale simulations. The main reactor parameters can be consequently determined based on more consistent calculations rather than on given assumptions. In the frame of this project different advanced models were developed to cover the major fusion technology areas, such as neutronics for the breeding blanket, electromagnetics and structural mechanics for toroidal and poloidal field coils and thermal-hydraulics for the balance of plant. Moreover, in order to prove the plant design so obtained from the physics standpoint, two important physics submodules were added and adapted to the project: the TREND and the TOKES codes, developed at Max-Planck-Institute for Plasma Physics Garching and KIT respectively. In this study the applied methodology is briefly described and the numerical results related to some improved reactor designs (e.g. based on current DEMO proposal) are reported and discussed. The main goal is to show the impact of accuracy and assumptions of the implemented models on main reactor’s parameters.
P3.200

Improved Solid Decomposition Algorithms for the CAD-to-MC Conversion Tool McCad

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McCad is a geometry conversion tool developed at the Karlsruhe Institute of Technology (KIT) for the automatic conversion of CAD models into the constructive solid geometry (CSG) representation. The resulting geometry models can then be used in Monte Carlo (MC) particle transport simulations applied in design analyses of fusion reactors like the DEMO tokamak developed within the European Power Plant Physics and Technology (PPPT) programme. The conversion of such a CAD model necessitates to decompose complex solids into a collection of disjoint and simple convex solids. The decomposition algorithm implemented previously in McCad turned out to be not very efficient and robust when applied to large and complex geometry models such as the DEMO tokamak. Frequently, programme crashes are encountered, irregular and fragmentized solids are produced, and a lot of CPU time and memory are consumed. To overcome such difficulties, new decomposition algorithms and functions have been developed and implemented in McCad. These include a new splitting surface generation algorithm for the decomposition of solids with curved surfaces, a new collision detecting algorithm using triangular facets, and a new splitting surface sorting algorithm based on feature recognition techniques. Furthermore, a new software architecture was introduced in McCad together with the new decomposition functions: These are implemented with an independent kernel module and thus can be integrated into any other CAD platform. This paper describes in detail the new decomposition algorithms, their implementation in McCad and the verification on the example of the European DEMO tokamak. The results show that this advanced McCad version is more efficient and robust and provides more accurate and less complex conversion results. It is thus concluded that the new McCad version is well suited for the conversion of highly complex tokamak models such as the European DEMO or ITER.
Primary Heat Transfer System design and safety evaluation for WCCB blanket sectors of CFETR

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The Water Cooled Ceramic Breeder (WCCB) blanket is one of the blanket candidates for Chinese Fusion Engineering Test Reactor (CFETR). In this work, the Primary Heat Transfer System (PHTS) of the WCCB blanket was designed based on the configuration of the blanket sectors, employing two identical loops at this stage. And each loop consists of a steam generator, a pressurizer and a main pump, feeding water coolant into 8 blanket sectors with blanket modules differing from one another. One of the loop was modeled using RELAP5/MOD3.3 with detailed blanket structures and heat sources in sectors, under normal condition and accidental scenarios. The operational mode of the PHTS was carefully chosen so as to obtain a more stable hydraulic behavior under steady state, due to the inhomogeneity of geometry structures and heat sources. Enveloping accidental cases, including in-vessel LOCA, in-box LOCA and ex-vessel LOCA, were selected to preliminarily evaluate the safety performance of the system. The results show the soundness of the system design.
P3.202

**Plasma protection module design of the ITER CIS fast architecture**

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The ITER Central Interlock System (CIS) architecture is composed of four categories of hardware: fast architecture, slow PLC based architecture, hardwired architecture and servers. The CIS fast architecture receives interlock events from various local plant systems of ITER and communicates the corresponding actions to any other local plant systems in order to avoid or mitigate the damage to the machine. Such functions require a reaction time that could range from 1 to 10 ms, which is faster than what the PLC are capable. The CIS fast architecture consists of a module named Plasma Protection Module (PPM), mainly in charge of interlock functions related to the plasma. As well as satisfying time performance requirements, the PPM complies with CIS reliability, availability and integrity requirements (probability of failure per hour below \(10^{-7}\) and failsafe solution). In this paper, we explain the engineering design of our approach under a technical perspective. A COTS FPGA in a redundant configuration solution, which uses serial communication with the local plant systems, is considered.
ITER is a prominent facility in the development of the nuclear fusion. It presents 44 ports providing access to the Vacuum Vessel at three different heights: Lower, Equatorial and Upper ports. Out of them, 22 ports, correspond to Diagnostics ports. They host a diversity of diagnostics systems, designed by the different ITER Domestics Agencies (DAs). They are later integrated into the different Diagnostics ports again by the DAs, not necessarily coincident with those who designed the systems. ITER Diagnostics ports represent a challenge from the neutronics standpoint. In addition to the general ITER port neutronics loads, Diagnostics ports present the difficulty of port plug penetrations with direct view to the plasma. Thus, Diagnostics ports design demands stringent nuclear analysis tasks, requiring large computational resources, long times and cumbersome methodologies and computational tools. Provided that many tenant will be implied in the nuclear analysis of each Diagnostics port, at different organization levels, the workload distribution procedure could easily lead to coordination problems: i) lack of standardization, ii) results spreading and iii) work replication. In order to avoid such problems, Diagnostics Division in IO-CT has developed MCNP models of the generic Diagnostics ports which are inserted into the latest ITER MCNP model C-lite version available. They will be used by all of the diagnostics tenants implied in the design of systems of port integrators. In this work, the MCNP models and their nuclear response during plasma operation are presented. This approach will serve to get three important objectives:

Establishing reference values to serve a basis for studies convergence and checking Models standardization, mandatory to avoid inconsistencies between works

Time, human and computational resources saving to all the implied tenants

Thus, reference nuclear analysis of ITER Diagnostics ports, 22 of 44 ITER ports, are presented in this work.
A novel approach to improve the calculation efficiency of the GVR method

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In order to control the global sample frequency, GVR method is deemed to be a practical way. But it is common that GVR method needs too many steps of weigh window iteration and it may fall into a long-history problem. We introduce a novel approach that is GVR method combined with reduced density in model, which could improve the calculation efficiency of GVR method in the following two aspects. One is that if the density were reduced by a reasonable factor, more neutrons would transport in the regions where far away from plasma. As a result, more flux results and more feasible GVR weight window would be obtained. It accelerates the iteration of GVR method by decreasing the steps of iteration. Secondly, using the reduced density weight windows could lead to smoother gradients in the neutron flux results. Thus, the gradients of weight windows are smoother accordingly. This is important as the very steep gradients in importance map that may lead to over splitting phenomenon or a long-history problem, which has detrimental effects on the parallel efficiency. Through some simulation experiments, this novel methods perform well, ensuring fewer steps of iteration and less possibility for long-history problem.
Hierarchical CAD-based modeling methods in SuperMC and its application in ITER C-lite model

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The advantages of CAD based Automatic Modeling make it possible to efficiently describe and verify complex nuclear system, such as ITER, for Nuclear Analysis. SuperMC/MCAM, the most widely applied CAD based Automatic Modeling tool for Monte Carlo, is currently focusing on modeling for Monte Carlo particle transport programs. Being more and more detailed, the radiation shielding modeling of fusion reactors has becoming more and more complex. For example, ITER C-lite has several more times of cells and surfaces than previous ITER reference neutronics models, which makes the CAD modeling time-consuming and difficult for error checking. In order to overcome this difficulty, A CAD-based hierarchical modeling method, named “CAD-based Hierarchical Modeling”, was developed for the purpose of managing geometries in hierarchical way, in which the whole transport space is subdivided into sub-regions accommodating corresponding sub-regions or geometric solids. It makes the overall CAD-based modeling being easy to break down to sub-tasks for both manual preprocessing and automatic conversion. In “CAD-based Hierarchical Modeling”, solids and sub-region of adjacent sub-regions may conflict with imperceptible overlaps which are acceptable in design accuracy but in the end may cause corruption of Monte Carlo particle transport calculation. An automatic overlapping eliminating method named “Threshold based Face Merging” was also developed to tackle this problem. During automatic conversion from CAD model to Monte Carlo model, it scans adjacent area between sub-regions, locate and remember the surfaces closer to each other than the threshold and merge them to a single one. Using the new methods, a SuperMC transport model of ITER C-lite was built according to the C-lite model released by ITER. The agreed calculation results between SuperMC on the newly built C-lite model and MCNP on the original C-lite model verified the correctness of the new methods.
Interfacing ATCA hot swap with PCIe hot plug for high-availability instrumentation in critical systems

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High availability (HA) is a key element in the specification of next generation Fusion devices, targeting steady-state operation. HA is especially required on mission-critical systems, as is the case of experimental Fusion devices and future Fusion power plants, where safety of people, environment and the infrastructure/investment is a primordial priority. IPFN developed control and data acquisition instrumentation under the PICMG 3.0 Advanced Telecommunications Computing Architecture (ATCA) specification, composed of digitizing and hub blades, which establish a PCIe data network through ATCA’s backplane Fabric Interface, connecting to an external host computer. The ATCA standard was selected to achieve a high level of availability, benefiting from its several redundancy resources, mandatory hardware management platform and native “Hot Swap” mechanism, which allows blade insertion and extraction without having to power off the system. From the host/software perspective, ATCA digitizers and hubs correspond to PCIe endpoint and bridge devices. The PCIe standard specifies a “hot plug” procedure for add and removal of these devices. Although PICMG has created an extension for the implementation of PCIe on the ATCA Fabric Interface, it does not specify hot swap implications on the PCIe hierarchy. On the other hand, PCIe hot plug states that for form-factors other than PCIe itself, the hot plug mechanism is implementation-dependent and should be defined by the form-factor. This paper describes the mechanisms, developed for an ATCA instrumentation platform, which will successfully coordinate hot swap mechanisms with the PCIe hot plug, in order to provide the overall system with HA. The present work also attempts to establish a relationship between these specifications, which could help to standardize PCIe hot plug implementation on the ATCA form-factor, fostering the development of instrumentation with such availability requirements.
Benchmarking of SuperMC activation calculation function and its application in ITER

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Activation study is very important for fusion reactors, from the view of component maintenance, occupational radiation exposure, and radioactive waste management. SuperMC is a multi-functional, intelligent, accurate and user-friendly simulation software system with comprehensive functions of transport simulation, material activation and transmutation, radiation source term and dose, etc. The activation calculation function of SuperMC was developed based on matrix exponential method - Chebyshev rational approximation method (CRAM). The users can specify the material composition and irradiation scenario with friendly user interface. The outputs include general properties, including the density, the composition, the number of stable isotopes, etc; specific activity; decay heat; contact dose rate; inhalation dose; ingestion dose; radioactive waste category; isotopic composition with pathway analysis; decay gamma spectrum. The program was tested with the handbook of activation data published by EURATOM/UKAEA Fusion and the international activation calculation benchmarks proposed by IAEA. The results show good agreements with the results from EASY. With supports from ITER International Organization, SuperMC was applied in ITER nuclear analysis. It was used to calculate the activation of bio-shield plugs and produce the activation handbook of ITER.
P4.001

Vacuum pumping system of neutral beam injector in KSTAR

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To achieve the high performance plasma in the Korea Superconducing Tokamak Advanced Research (KSTAR) tokamak, Neutral Beam Injection (NBI) system has been installed and upgraded. The first NBI (NBI-1) was installed in 2010, which provides a 100 keV deuterium neutral beam of 6 MW maximum using three ion sources. The second NBI (NBI-2) with another 6 MW will complete to be constructed by 2018. As the vacuum can have substantial effect on the beam performance, the design of the Vacuum Pumping System (VPS) for NBI-2 has been carefully conducted. Accordingly, the operation results of the VPS for NBI-1 was analyzed and feedbacked to the design of NBI-2. In particular, the cryosorption pump, which is used to exhaust the massive gas inrushing from ion sources and a neutralizer, emerged as a major design issue. In this paper, the design of cryosorption pumps for NBI-2 were derived and the overall vacuum of NBI-2 was analyzed including turbo pumps, mechanical booster pumps, and dry pumps.
Development of the new KSTAR helium distribution box

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KSTAR project has required the new helium distribution box named upgraded distribution box (DBU) for the operation of the cryogenic components such as in-vessel cryo-pump (CPI), super-sonic molecular beam injector (SMBI), and hydrogen pellet injection system (PIS). Two CPIs are inserted into the tokamak vacuum vessel and these components shall be operated at 90 K for the liquid nitrogen thermal shields and 4.5 K for the hydrogen cryo-panel. One hydrogen PIS shall be mounted to the tokamak for the 2016 KSTAR campaign. Liquid nitrogen shall be supplied to the one SMBI. For the operation of above mentioned 3 kinds of cryogenic components, a helium refrigerator, which had been used for the R&D in the KSTAR facility construction phase (2002 ˜ 2013), was moved and inserted into the KSTAR 9 kW helium facility room. The cooling capacity of the refrigerator at 4.5 K is 1 kW and it was manufactured from the Linde Kryotechnik before 2002. From the beginning of 2015, design and fabrication of the DBU was started. It shall control the liquid nitrogen for the SMBI and CPI thermal shields whereas liquid helium for the CPI cryo-panel and PIS. To minimize the temperature of the liquid nitrogen to be supplied to SMBI and CPI, a thermal damper tank was inserted into the distribution box. Nitrogen return gases are to be warmed up to room temperature at the heater in the distribution box. A 1000 liters of liquid helium vessel is located nearby the PIS to supply cold gas helium (˜ 5 K). Because the CPI cryo-panel requires regeneration up to 90 K, complex regeneration and re-cool down scenario was developed and applied to the DBU. Including operational results, details of the DBU progresses will be reported in this paper.
The nuclear fusion research is in progress for the next generation energy source in many countries. The Korea Superconducting Tokamak Advanced Research (KSTAR) in Korea, the Experimental Advanced Superconducting Tokamak (EAST) in China and the Wendelstein7-X in German are the operational superconducting fusion device in the world. The International Thermonuclear Experimental Reactor (ITER) is in construction under the international collaboration between China, EU, India, Japan, Korea, Russia and the USA and it will be completed the construction until the mid of 2020’s. Above listed fusion devices are consisted of low temperature superconductor and have to be operated at the liquid helium temperature. To achieve cryogenic temperature for the superconducting fusion device, the large cryogenic helium plant is necessary and the efficient cool-down procedure should be required. The KSTAR device is fully superconducting (SC) tokamak that consists of 16 Toroidal Field (TF) magnets and 14 Poloidal Field (PF) magnets. The KSTAR SC magnets are made of Nb3Sn and NbTi and its cold mass is around 30 tons. A helium refrigeration system (HRS) with 9 kW @ 4.5K had been installed to keep the KSTAR SC magnets at appropriate temperature condition and it has been operated successfully since 2008. As a result of cool-down optimization by the KSTAR cryogenics personnel based on the operational experience, the elapsed time for cool-down of the KSTAR SC magnets has been reduced. In this paper, the result of KSTAR cool-down and warm-up since 2009 will be presented and the operation parameter of the KSTAR HRS during the cool-down and warm-up will be analyzed and discussed.
Issues of arc discharge for long pulse KSTAR NBI

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In 2015 KSTAR Campaign, the maximum injection power of the KSTAR tangential Neutral Beam Injector (KSTAR NBI-1) is 5.39MW with three ion sources. Issues in beam extraction found during the experiment were 1) a large oscillation of beam current, 2) frequent interrupts in beam extraction due to breakdown in grids, and 3) a distortion of waveform. To solve these issues, we focused on the unstable arc discharge. Depending on the control mode and plasma impedance, characteristics, current overshoot and oscillation, of Arc voltage/current waveforms are affected. Because of initial waveform distortion, continuous interruption on beam extraction prevents stable operation. Constant-Power control mode (CP mode) results in severe overshoot and oscillation having amplitude of 10V. Adjusting interval between pre-arc discharge and main-arc discharge, or the duration of pre-arc discharge was not effective. Applying slow rising time (20ms) of CP mode or the combination of initial Constant-Voltage control mode (CV mode) and CP mode also causes distortion and overshoot. Optimized control gains with CP mode without intentional ramping applied to dummy load results 1) reduced oscillation amplitude of about 4V, 2) stable initial rising in 4ms, and 3) elimination of distortion with optimized notch resister. Performance of power supply with arc discharge is also discussed.
Development of pellet injection system for KSTAR

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KSTAR (Korea Superconducting Tokamak Advanced Research) has used gas puffing system as main fueling method since 2008. Up to date total fueling efficiency of gas puff is less than 30%. Pellet injection is more effective technique to control plasma density than gas puffing system and supersonic molecular beam injection. Many fusion devices such as JET, Tore Supra, ASDEX-U, HL-2A, EAST, and LHD have already installed the pellet injection system (PIS) and then have shown the impressive results of plasma density control and ELM (Edge Localized Mode) mitigation. Pellet injection system for KSTAR consists of pellet injector which injects pellets of hydrogen isotope (diameter of 2 mm, length of 1.5 to 2 mm) at speed above 200 m/s with injection frequency of 1~20 Hz, 2 stage differential pumping system, guide tube, control system and so on. It is planned to inject pellets through high field side and increase plasma performance. The specification of KSTAR pellet injection system, design of differential pumping system and guide tube, as well as configuration of control system together with test results achieved in 2016 are presented.
Final design and prototyping of the SPIDER caesium ovens

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The Padova Research on ITER Megavolt Accelerator (PRIMA), under construction at Consorzio RFX, will host SPIDER test bed, a full-size 100 kV negative ion source, and MITICA test bed, a prototype of the whole ITER injector, aiming to develop and optimize the heating injectors to be installed in ITER. The production of hydrogen (or deuterium) negative ions inside the sources relies mainly on the generation on a caesiated surface with a low work function, which enhances the ion yield and reduces the co-extracted electrons. In the SPIDER beam source fresh Cs has to be injected in a controlled way by means of three Cs ovens. Since the Cs ovens are embedded in the source, their design has to assure vacuum and high temperature compatibility, and the remote operation. The design is complicated by the high reactivity of Cs, the difficulty in setting a stable (but adjustable) temperature distribution among the parts under vacuum, and need for the development of a special solenoid valve. The final design, which concludes an initial design and R&D phase, is carried out with the IPP Garching collaboration and the experience gained at ELISE test bed. Specific diagnostics are embedded to measure the Cs vapour flux exiting from the injection nozzle, by means of a Surface Ionization Detector (SID) developed at IPP, and also the amount of Cs in the reservoir, with a level indicator. The procurement of the three ovens is preceded by the setting up of a Cs oven test bed, where a prototype of oven is tested and characterised inside a vacuum chamber. The final design of the oven, also supported by finite element thermal analyses, and the R&D activity on the solenoid valve are presented. The Cs test bed is described and the results of the prototyping activities are given.
The heating neutral beam injectors (HNBs) at ITER are expected to deliver 33 MW of neutral beam power to the ITER plasma for the purposes of heating and current drive. This is achieved by using 2 injectors, each capable of delivering 16.5 MW of neutral beam power. The beam source of each injector is a complex assembly composed by an RF based negative ion source having an extraction area of \( \sim 1.6 \, \text{m}^2 \) and a 1 MV accelerator having 5 stages of 200 kV each. As a similar source meeting the ITER beam specifications is yet to be realized and the functional requirements have never been met, it was recognized as necessary to setup a test facility, PRIMA (Padova Research on ITER Megavolt Accelerator), in Italy, including a full-size 100 kV negative ion source, SPIDER test bed, and a prototype of the ITER HNB injectors, 1 MV MITICA test bed. The SPIDER beam source procurement started in October 2012 and is expected to be delivered during the second half 2016. Some technical challenges have been overcome during the manufacturing phase and prototypes have been developed to qualify some design details and manufacturing processes for different source components. All the parts and components of the beam source are currently manufactured and tested. The next phase of assembly at the factory is expected to begin in the second quarter of 2016 and requires a well-developed sequence of operations not only to assemble the different components of the source ensuring electrical insulation, leak tightness and grids alignment within the envisaged tolerances but also of the mounting of various diagnostics, which will aid the operation and optimization of the source parameters. The paper will highlight procurement challenges, the technical achievements and the assembly experiences of this complex component.
The ITER project requires at least two Neutral Beam Injectors, each accelerating to 1MV a 40A beam of negative deuterium ions, to deliver to the plasma a power of about 33 MW for one hour as additional heating. Since these requirements have never been experimentally met, it was recognized necessary to build-up a test facility, named PRIMA (Padova Research on ITER Megavolt Accelerator), in Italy, which includes both a full-size negative ion source (SPIDER - Source for Production of Ion of Deuterium Extracted from RF plasma) and a prototype of the whole ITER injector (MITICA - Megavolt ITER Injector & Concept Advancement). This realization is made with the main contribution of the European Union, through Fusion For Energy, the ITER Organization and Consorzio RFX which hosts the Test Facility. 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, current uniformity and duration. The paper describes the Gas injection and Vacuum System (GVS) starting from the analysis of the requirements and going through the detailed design and the procurement of the system up to the site acceptance tests. In particular the rationale behind the main design choices are presented and some manufacturing details of the gas injection plant feeding the RF source is given. Furthermore, the sensor system dedicated to the measurement of the vacuum level, gas pressure and throughput and the residual gas analysis are described considering the interfaces with the PRIMA interlock and safety system. Reference is also made to safety aspects concerning the presence of H2/D2 in a closed environment as the SPIDER bioshield is.
P4.010

Speeding up predictive electromagnetic simulations for tokamak application

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Large state-of-the-art fusion devices involve extensive computations throughout the engineering design process from the concept to the commissioning. A variety of well-established software tools, such as ANSYS, OPERA, CARIDDY, TYPHOON, TORNADO has produced a range of simulation techniques and approaches for electro-magnetic (EM) simulations of principal components of tokamaks. The installation activities make it possible to adjust computational models using inspection and measurement data. A computational technique is presented that is targeted to speed up parametric and predictive EM simulations with intensive data communication. A large number of parameters on various scenarios should be efficiently correlated before, during and after experiments on fusion devices in order to generate a consistent operational database. The paper presents an attempt to proceed to a general concept of software environment for fast and consistent multi-task simulation of EM transients (engineering simulator for tokamak applications). The strategy exploits parallel processing with optimized simulation algorithms, based on using of influence functions and superposition principle, and improved intertask communication to take full advantage of parallelism. The software has been tested on a multi-core supercomputer. The results were compared with data obtained in standard TYPHOON computations. A discrepancy was found to be below 0.4%. The computation cost for the simulator is proportional to the number of observation points. An average computation time with the simulator is found to be by hundreds times less than the time required to solve numerically a relevant system of differential equations for known software tools.
Systems code studies on the optimisation of design parameters for a DEMO tokamak reactor

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In the European strategy towards fusion electricity, a demonstration tokamak fusion reactor (DEMO) is foreseen as the single step between ITER and a fusion power plant. Recent studies have been focussing on the concept development for a “conservative” pulsed tokamak reactor with an electrical output power of $P_{e}\sim 500$ MW and plasma pulse duration of $t_{\text{pulse}} \sim 2$ hours. In the design process for DEMO, systems codes are commonly used as numerical tools for optimisation studies. Systems codes use a simplified but comprehensive description of the most important features of the plasma and of the main technical elements of the reactor, together with a number of boundary conditions and limitations from physics and technology, which altogether represent the current status of knowledge. The key performance data of the reactor such as electrical output power and plasma pulse duration are depending on a variety of design and plasma parameters such as plasma minor and major radius and shaping (elongation and triangularity), plasma density, safety factor, energy and particle confinement quality, auxiliary heating, impurity concentrations (plasma radiation), magnetic field strength and coil geometry, radial build (blanket thickness) and permissible wall and divertor loads. Within this multi-dimensional parameter space, different approaches can be used to define the goals of optimisation. Within this paper, a systems code has been used to perform a variation of magnetic field, aspect ratio and confinement quality, for the case of tokamak reactors with $P_{e}\sim 500$ MW, to evaluate the effects on investment cost, plasma duration, divertor loads and other quantities, as compared to a reference case with $t_{\text{pulse}} \sim 2$ hours. It is shown that, compared to the reference case, reactor designs with reduced divertor loads and significantly increased pulse duration can be achieved when accepting a moderate increase of investment cost.
Speeding up predictive electromagnetic simulations for tokamak application

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Large state-of-the-art fusion devices involve extensive computations throughout the engineering design process from the concept to the commissioning. A variety of well-established software tools, such as ANSYS, OPERA, CARIDDY, TYPHOON, TORNADO has produced a range of simulation techniques and approaches for electro-magnetic (EM) simulations of principal components of tokamaks. The installation activities make it possible to adjust computational models using inspection and measurement data. A computational technique is presented that is targeted to speed up parametric and predictive EM simulations with intensive data communication. A large number of parameters on various scenarios should be efficiently correlated before, during and after experiments on fusion devices in order to generate a consistent operational database. The paper presents an attempt to proceed to a general concept of software environment for fast and consistent multi-task simulation of EM transients (engineering simulator for tokamak applications). The strategy exploits parallel processing with optimized simulation algorithms, based on using of influence functions and superposition principle, and improved intertask communication to take full advantage of parallelism. The software has been tested on a multi-core supercomputer. The results were compared with data obtained in standard TYPHOON computations. A discrepancy was found to be below 0.4%. The computation cost for the simulator is proportional to the number of observation points. An average computation time with the simulator is found to be by hundreds times less than the time required to solve numerically a relevant system of differential equations for known software tools.
Systems code studies on the optimisation of design parameters for a DEMO tokamak reactor

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In the European strategy towards fusion electricity, a demonstration tokamak fusion reactor (DEMO) is foreseen as the single step between ITER and a fusion power plant. Recent studies have been focussing on the concept development for a “conservative” pulsed tokamak reactor with an electrical output power of $P_{el} \sim 500$ MW and plasma pulse duration of $t_{pulse} \sim 2$ hours. In the design process for DEMO, systems codes are commonly used as numerical tools for optimisation studies. Systems codes use a simplified but comprehensive description of the most important features of the plasma and of the main technical elements of the reactor, together with a number of boundary conditions and limitations from physics and technology, which altogether represent the current status of knowledge. The key performance data of the reactor such as electrical output power and plasma pulse duration are depending on a variety of design and plasma parameters such as plasma minor and major radius and shaping (elongation and triangularity), plasma density, safety factor, energy and particle confinement quality, auxiliary heating, impurity concentrations (plasma radiation), magnetic field strength and coil geometry, radial build (blanket thickness) and permissible wall and divertor loads. Within this multi-dimensional parameter space, different approaches can be used to define the goals of optimisation. Within this paper, a systems code has been used to perform a variation of magnetic field, aspect ratio and confinement quality, for the case of tokamak reactors with $P_{el} \sim 500$ MW, to evaluate the effects on investment cost, plasma duration, divertor loads and other quantities, as compared to a reference case with $t_{pulse} \sim 2$ hours. It is shown that, compared to the reference case, reactor designs with reduced divertor loads and significantly increased pulse duration can be achieved when accepting a moderate increase of investment cost.
Modelling magnetic effects for steel rebar of tokamak buildings

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Concrete structures of tokamak buildings are reinforced with steel rebar that produces a substantial contribution into the tokamak field both in the plasma region and in the building where the service staff and magnetically sensitive equipment will be located. The article describes an advanced approach to modelling magnetic properties of reinforced concrete structures bearing in mind the anisotropic effect. Analytical and numerical evaluations have been obtained for equivalent anisotropic properties of the rebar pattern with respect to its realistic geometry and permeability. The equivalent model has been validated in the computation of a test problem. For comparison, simulations have been carried out with a detailed 3D FE model that describes each of the steel rods. For the simulations the codes KOMPOT and KLONDIKE have been applied. The equivalent model has required about ten time less finite elements than the detailed model. A comparison of the fields obtained has demonstrated a very good match, even for the distances comparable with the rebar rod spacings.
Control strategy for mitigation of pulsed heat power transferred to cryoplant from superconducting magnets

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Cryogenic systems for fusion reactors have to cope with large pulsed heat load generated during fusion experiments. The paper is focused on mitigation of pulsed heat power arriving to the cryoplant from several parallel cooling loops of tokamak superconducting magnets. A new control strategy is proposed. The pressure drop measured at the return cryoline serves as a feedback signal to mitigate the total heat load on the cryoplant by adjustment of a circulator speed. The efficiency of the strategy has been numerically tested with the thermal-hydraulic code Venecia using a model of the ITER tokamak TF coils with the support structures and the cryodistribution system. Results of the simulations are presented for the normal operation and for the plasma disruption event followed by fast discharge.
Fault energy dump & repeated breakdown validations of HVPS commissioned for negative ion source-ROBIN

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Successful operation of a Neutral Beam Injector is dependent on the performance of High voltage power supply system (HVPS) for the production of ion beam. To meet the functional requirements of ion extraction, the power supplies (PS) are designed for fast output cut-off, low energy content during breakdown (BD), ability to withstand repeated BD. It is important that features of the PS are validated in a manner that ensures their integration with ion source. For negative ion source ROBIN, the HVPS consists of two DC PS, namely extraction power supply system (EPSS: -11kV, 35A) for the extraction of negative ions, which floats over Acceleration power supply system (APSS: -35kV, 15A) for the acceleration of negative ion beam. The following special attributes of the PS and their qualification procedure have been carried out, prior to their integration with ROBIN: 

a) Energy storage- The PS have been designed with fast output cut-off (<10μs) to ensure the energy dump during BD is <10J to avoid damage to the grids of the ROBIN. A wire test has validated this critical feature; 

b) BD and subsequent reapplication of HV for 200 times- The test for the repeated BD has been carried out on each power supply individually and also with EPSS floating over APSS, up to 200 times with 15ms HV inhibition between BD. The peak current observed during BD is ~250A. A passive BD protection scheme has been implemented to further reduce the peak current during BD. While conducting these tests, critical technical issues like voltage shoot-up across EPSS during BD in APSS, noise interferences during BD etc. were encountered which seriously affected the performance of the PS. These issues were resolved after investigating their causes and implementing the appropriate solution. The paper will highlight about testing methodology for validation of fault energy dump and repeated BDs, passive BD protection scheme and issues encountered during testing & their resolution.
Institute for Plasma Research (IPR), India has a programme of development of allied technologies with applications related to fusion reactor. A pneumatic gas gun kind Single pellet injector system (SPINS-IN) developed at IPR is successfully delivering hydrogen pellets of size 2 mm with a velocity of 700 meters/sec. It is a cryocooler based system operated at a temperature < 10 K and delivering a pellet every five minutes equipped with a fast opening valve for pellet acceleration and necessary diagnostic to measure the ejected pellet parameter. Its differential pumping system comprises three vacuum chambers to remove the propellant helium from injection line. Gas content of 2mm to 3mm pellet is of the order of 7 to 23.3 mbar-liter. For a plasma temperature in the range of 1 to 3 KeV and density $5 \times 10^{19} \text{cm}^{-3}$, a study was carried out using NGS model for penetration depth of pellet in plasma. Injector is now installed on SST-1 tokomak for pellet injection related experiments. For continuous supply of pellets development of Extruder Type Pellet Injector System (ETPIS) is in progress. It is a twin screw based cryogenic extruder with a precooler and liquefier. Cool down will be carried out using cryocooler and the system will extrude hydrogen ribbon. This paper describes the progress of pellet injector technology development at IPR-India.
Developments in pellet injection technology for transient mitigation applications

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The formation and acceleration of cryogenically solidified pellets of hydrogen isotopes has long been under development for fueling fusion plasmas. Fueling with DT pellets injected from the high field side wall has been proposed for future burning plasma tokamak devices. In addition to fueling, smaller shallow penetrating pellets of deuterium injected from the low field side wall have been shown to be capable of triggering rapid small edge localized modes (ELMs) to limit heat flux damage from large naturally occurring ELMs. Another application of pellet injection is that of disruption mitigation where large pellets of neon, argon, and deuterium mixtures are produced and shattered upon injection into disrupting plasmas to quickly radiate the plasma energy in order to mitigate possible damage to in-vessel components. The pellets for fueling and ELM triggering need to be formed continuously for fusion applications. Screw extruder systems are under development that can produce either fueling size pellets or ELM triggering pellets in steady state. The required throughput for future burning plasma devices is well beyond that for present day devices and requires stable cooling from a super critical helium source. The large shattered pellets for disruption mitigation are formed in-situ in a pipe gun device and held intact until needed during a disruption. Pressurized gas is also used to accelerate these pellets, but the gas enters the vessel and is not captured. Prototypes of pellet injectors for these applications have been developed and tested in the laboratory and deployed on present day experiments. Details of these injector designs and their applicability for future burning plasma devices will be presented.
P4.016

Ignitor siting at the TRINITI site in Russian Federation

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The compact, high field fusion experiment Ignitor aims at the demonstration, for the first time, of ignition in magnetically confined D-T plasmas, together with the exploration of the physics of the ignition process, and of heating and control of plasmas under controlled burning conditions. The machine parameters have been established on the basis of existing knowledge of the confinement properties of high density plasmas and technological feasibility. The collaboration between Italy and Russia is centered on the construction of Ignitor in Italy and its installation and operation within the Troitskii Institute of Innovative and Thermonuclear Research (TRINITI) site (near Troitsk, Greater Moscow, Russia). The site, hosting since decades nuclear installations, is well characterized, both from the meteorological and population aspects: large amounts of data were collected over the years. Moreover, many assessments made for the SFT (Strong Field Tokamak) are applicable to Ignitor. A Tritium system to be operated up to 10g/day is already available, much more than the Ignitor requirements. Being TRINITI a fully characterized nuclear site, we can conclude that it is fully apt to host Ignitor, according both to international and to Russia’s regulations. A Preliminary Safety Report (PSR) in under preparation for the siting of Ignitor at TRINITI. Deterministic evaluations (radioactive inventory and dose population codes) have been performed. The preliminary radiological impact analysis for the normal operation and the main accidental sequences of Ignitor, for the case of its localization at TRINITI are presented, along the lines of the assessment done for the Italian site of Caorso. The Ignitor machine, both during routine functioning and accidental sequences, presents a negligible radiological impact. No need of people evacuation or any emergency countermeasure is necessary even in presence of the worst accident.
Feasibility study for an advanced nuclear fusion experiment
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Nuclear Fusion is a candidate as a long-term energy solution for developed countries. A fusion plasma can be fuelled by different kinds of isotopes. The advantages of Deuterium-Helium-3 (DHe) plasmas of advanced fusion reactors lie in the scarcity of neutrons (due to side DD and DT reactions), and direct conversion of the produced energy without thermal cycle. The proposed CANDOR DHe plasma experiment design, based on the technologies developed for Ignitor, has been recently reviewed. The design evolution, called Candor-II, is presented: it should be able to reach fusion conditions on a DHe plasma based on existing technologies and knowledge. The new device is intended to be the cornerstone for studies of DHe plasmas: its dimensions are larger than Ignitor, although still compact and based on the same operating components. Detailed investigations by NASA show that obtaining He3 from the moon surface is technically feasible and economically viable. The annual fuel needs for Candor-2 would amount in around 2 g of He3, that is, around 60 g for all its operating life. Lunar mobile mining techniques, with the concept of a mobile miner, were developed by NASA. The net annual collection rate of such a unit is 33 kg of He3. Concerning costs, the total cost of extraction of He3, all inclusive from lunar mining down to He3 transportation to Earth is estimated to be 1000 USD/g Therefore, the fuel cost for Candor would be around 60,000 USD in 30 years, or around 2000 USD per year. Our study on DHe compact high-magnetic field tokamaks shows that no environmental problems arise from such devices, from the radiological point of view. The DHe fusion cycle offers strong safety advantages and could be the ultimate response to the environmental requirements for future nuclear power plants.
Design of rotating resonant magnetic perturbation coil system in the STOR-M tokamak

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The interaction between resonant magnetic perturbations (RMP) and plasma is an active topic in the fusion energy research. RMP involves the use of radial magnetic fields generated by external coils installed on a tokamak device. The resonant interaction between the plasma and the RMP fields has many favorable effects such as suppression of instabilities and improvement of discharge parameters in tokamaks. The RMP technique has been successfully implemented in the STOR-M tokamak. A set of \((m = 2, n = 1)\) helical coils carrying a current pulse was used to study the effects of RMP on magnetic islands, plasma rotation, and other edge plasma parameters. The current RMP coil system creates only a stationary magnetic field that does not rotate with the magnetic islands. A new design of RMP system is being developed for the STOR-M tokamak. The system consists of a number of external saddle coils distributed in the poloidal and toroidal directions. The saddle coils will be powered by AC power supply to generate a rotating RMP field. The advantage of producing a rotating RMP with variable phase and frequency is the possibility to stabilize the targeted magnetic islands without mode locking which is a major cause for plasma disruptions. Numerical simulations have been carried out to calculate several parameters for the new RMP system such as the self-inductance of the saddle coils, the magnetic field generated by the coils, as well as the dominant modes. The dominant mode generated by the new RMP coil system may be tuned to \((2, 1)\) with a significant contribution from \((2, 3)\) and \((2, 5)\) modes.
EU DEMO transient phases: main constraints and heating mix studies for ramp-up and ramp-down

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EU DEMO studies for pulsed (DEMO1) and steady-state (DEMO2) concepts are currently in the pre-conceptual phase [1]. DEMO1 aims at producing about 2GW of fusion power with a burn time of approximately 2 hours. Within EUROfusion Power Plant Physics and Technology department, DEMO scenario modelling is carried out as part of the validation of feasibility and performance of DEMO designs. One of the most challenging activities deals with numerical investigations of DEMO1 transient phases including ramp-up and ramp-down, which exhibit peculiar issues with respect to existing devices. Studies on ramp-up have been carried out to highlight the effects of different ramp-up options in terms of robustness of the access to the desired flattop scenario. A heating power during ramp-up, additional to the one required during flattop, appears to be necessary for plasma burn initiation and access to H-mode, with $P_{aux,RU} \geq 50$MW depending on the uncertainties on L-H transition scaling. Current ramp-rate and heating power influence also plasma position controllability, and results are presented in terms of the achieved $\ell_i(3)$ and $\beta_p$.

Additional power requirements and integration of different systems which are relevant for DEMO heating mix assessment are here discussed. Ramp-down phase in DEMO poses specific issues on vertical stability given the distance of control actuators from the plasma. Ramp-down trajectories with controllable plasma boundaries have been coupled to transport studies showing the necessity of additional ramp-down heating power to avoid radiative plasma collapses. Off-axis power deposition helps plasma controllability, together with a current ramp-rate $\leq 100$kA/s. Plasma radiation also dominates the H-L transition, which is investigated and appears to be a critical step in terms of plasma control. DEMO performance is strongly linked to the maximum plasma elongation, which has to be assessed comparing different ramp-down trajectories. [1] G. Federici et. al, Fus. Eng. Des. 89, 882-889 (2014)
The Heating & Current Drive (H&CD) systems in a DEMOnstration fusion power plant are one of the major energy consumers. Due to its high demand in electrical energy produced in the balance of plant (BoP) the H&CD efficiency optimization is one of the main goals of the DEMO development. The energy consumption of the H&CD sub-systems in different plant modes & states and plasma phases need to be strongly considered for the DEMO conceptual design. The H&CD power for DEMO, based on physics scenarios for the plasma phases, is needed for plasma breakdown, plasma initiation, heating to H-mode, burn control, controlled current ramp-up and -down, MHD control and other functions. Plasma control will need significant installed HCD power, though not continuously used. Previously, in the DEMO baseline definition, optimistic forecasted H&CD efficiencies had been assumed in the corresponding system code (i.e. PROCESS) module. Realizing that there is a high uncertainty in the assumptions, hence to move closer to a mature design, it is proposed to use more realistic state-of-the-art efficiencies. Those designs must have achieved a minimum Technical Readiness Level (TRL) either by evidence and validation in the laboratory or preferably by being tested in a relevant environment. This presentation discusses the transition from previous to present assumptions and the impact on the DEMO power plant and basic tokamak configuration. A comparison of the various HCD systems NBI (Neutral Beam Injection), Electron Cyclotron (EC), Ion Cyclotron (IC) in terms of impact on Tritium Breeding Ratio (TBR) due to various openings for the H&CD in the breeding blanket (BB) is presented. For increasing the reliability as major features the power per system unit and the redundancy are identified leading to a new proposal for clusters for EC and modular ion-sources for NB.
Multi-design innovative cooling research & optimization (MICRO): novel proposals for high performance cooling in DEMO

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Several novel design solutions for high performance cooling systems have been developed by Consorzio RFX, permitting to experimentally simulate the challenging heat transfer conditions foreseen in the future fusion devices. The project, called Multi-design Innovative Cooling Research & Optimization (MICRO), aims on one hand to verify the present solution applied inside the MITICA experiment and on the other to perform further improvements with an acceptable pressure drop and reliable manufacturing process. A comprehensive parametric investigation has been carried out with the goal of comparing various design options and establishing a standard approach to be applied in several devices, characterized by comparable heat loads both in terms of spatial distribution and amplitude. The main advantages rely on the possibility to extend the fatigue life-cycle of different high thermal stress components and to investigate the possibility to employ alternative dielectric fluids instead of water. Design solutions with an intrinsic enhancement of heat transfer process would in fact allow the exploitation of less performing fluids in terms of cooling capability. However, if the unavoidable deterioration of the cooling parameters would not prevent to satisfy the thermo-structural requirements set for such kind of components, these dielectric fluids would represent a significantly advantageous option with respect to the existing technologies. This is particularly relevant in view of DEMO and future power plants characterized by higher efficiency and reliability. This paper gives a detailed description of the Computation Fluid Dynamics (CFD) analysis, of the samples manufacturing and of the experimental tests that have been carried out so far. 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.
Numerical and semianalytical treatments of neutral beam current drive in DEMO-FNS

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Neutral Beam Current Drive (NBCD) is considered as an indispensable mechanism for a steady state regime in such contemporary projects as a tokamak based neutron source or a DEMO type thermonuclear reactor. In this report numerical calculations of NBCD with a Monte Carlo code NUBEAM are complemented by a semianalytical treatment of fast ion velocity distribution function. NBCD parameters were obtained for DEMO-FNS project varying the neutral beam injection geometry as well as the major radius of a device. Specific features of NBCD for a low aspect ratio configuration are illustrated by applying our approaches to Spherical Tokamak Neutron Source FNS-ST. A detailed comparison of numerical and semianalytical approaches is presented on the basis of ASTRA transport code provided equilibrium data for different plasma regimes.
Progress on the ENEA 500 kW, 250 GHz CARM design

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F. Mirizzi1, M. Carpanese2, S. Ceccuzzi2, F. Ciocci2, G. Dattoli2, E. Di Palma2, A. Doria2, G.P. Gallerano2, G. Maffia2, A. Petralia2, G.L. Ravera2, E. Sabia3, I. Spassovsky2, A.A. Tuccillo2, S. Turtù2, P. Zito2 on behalf of the ENEA CARM task force. 1Consorzio CREATE, Via Claudio 21, I-80125 Napoli, Italy 2ENEA, C.R. Frascati, via E. Fermi 45, 00044 Frascati (Roma), Italy An ENEA task force is studying and designing a Cyclotron Auto Resonance Maser (CARM), with a RF power in excess of 500 kW and a nominal output frequency of 250 GHz. Its main application is foreseen in the ECCD system for DEMO, which expected very high electron temperatures call for EC frequencies in the 200 – 300 GHz range. Two different steps are foreseen before the realization of the final full specs CARM, the first one dedicated to the optimization of a 50 µs, 100 kW prototype. The second step foresees the test on FTU of a 500 kW prototype with a pulse length up to 100 ms. A really intense analytical work has been performed by the task force allowing the definition of the main parameters of the CARM, like beam voltage and current, magnetic field intensity and profile. The study of the high voltage power supply and modulation system for the first step of has been completed and the overall electrical parameters of this system are already available. An Electron gun and Electrodynamical system are designed too. The available results of the CARM analysis and design are presented in the paper.
Traditional vs. advanced Bragg reflectors for oversized circular waveguide

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In the frame of the feasibility study of a Cyclotron Auto-Resonance Maser (CARM), different solutions for the distributed reflectors of the resonant cavity have been considered and compared. In detail, a 250 GHz CARM source is under design with an output power of 200 kW for pulses up to 0.2 s, representing the first milestone of a more ambitious project, aimed at achieving a CW 1 MW mm-wave generator. Such devices are potentially very attractive as sources in the electron cyclotron range of frequencies for reactor-relevant magnetic-confinement fusion machines like DEMO. The CARM cavity is a highly oversized smooth-wall circular waveguide sandwiched between two Bragg reflectors: the one (upstream mirror) at the gun side, the other (downstream mirror) at the window side. The former is challenging because reflectivity in excess of 95% for the working TE53 mode requires more than 700 corrugations. Accordingly alternative solutions to the traditional bandgap-based mirror, like advanced and tapered Bragg reflectors, have been investigated in terms of reflectivity, bandwidth, number of ripples and conductor losses. Strengths and weaknesses of each mirror are compared, showing that advanced Bragg reflectors lead to shorter devices with higher ohmic dissipation and smaller bandwidth. The study has been carried out with a mode-matching in-house code, developed on purpose, where accurate calculation of ohmic losses is implemented. To the best of authors’ knowledge, this is the first comprehensive comparison of such overmoded components entirely based on full-wave results.
Integrating a distributed antenna in DEMO: requirements and challenges

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The use of efficient heating and current drive systems is an important research priority for DEMO. The Ion Cyclotron Resonance Heating (ICRH) is one such system justified by its inherent advantages, though in its present status (antenna situated in a port in the Vacuum Vessel (VV)) is unacceptable for DEMO, where tritium self-sufficiency is to be demonstrated, and reducing the openings in the VV is essential (since they contribute to tritium breeding). To address these issues, a novel ICRH concept is currently under development, consisting of a toroidally continuous antenna, integrated in the First Wall (FW). Such configuration calls for a strong machine integration; it is therefore important that the ICRH antenna is considered from the beginning in the machine design. Major engineering constraints are imposed by the machine, including the hosting blanket modules, where the antenna will have e.g. to not impair the blanket functions (including shielding and tritium breeding) and have the same level of operational safety as the FW, but also by the Remote Handling (RH) process, where the antenna and feeding lines will have to be integrated in a way so that no complexity in the RH procedure is added. This abstract describes the most up to date engineering constraints imposed by the present DEMO configuration, serving as guidance for the ongoing antenna design process. 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.
Development of an arc ion source based on Marx generator for VEST NBI system

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A high-power pulsed arc ion source based on Marx generator has been developed at the Korea Atomic Energy Research Institute for the heating NBI system of the VEST which is a compact spherical tokamak at Seoul National University to study the reactor-relevant tokamak operating scenario[1]. The NBI system, with a total ion beam power of 0.8MW, was designed for the core plasma heating. However, the beam injection energy is required to be lower than 10 keV under the initial target plasma parameters of VEST due to beam-plasma coupling efficiency. For this reason, the beam injection energy needs to be modulated in a pulse duration of below 20 ms. To satisfy these requirements, the pulse power system based on a two-stage Marx generator is designed by utilizing high-energy capacitors and a solid-state switching system for the ion beam extraction and arc power supply. In addition, a battery-based power supply system is designed for the filament heating power supply. By using the pulse power supply system, large area pulsed arc ion source is successfully commissioned. Plasma parameters of the source in the pulse duration are measured by triple probe diagnostic and characteristics of the extracted ion beam is determined under various operating conditions. References: K. J. Chung, Y. H. An, B. K. Jung, H. Y. Lee, C. Sung, Y. S. Na, et al., Plasma Science and Technology 15 (2013) 244
P4.027

Design of an impedance matching circuit for a high power rectangular RF ion source

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Within the framework for development of the radio frequency (RF) driven positive ion source as an alternative to the conventional filament arc driven ion source for fusion applications, KAERI is currently constructing a new high power (50 kW at a frequency of 2 MHz) large area RF ion source. The ion source was designed to have a rounded rectangular geometry for covering rectangular ion extraction area of 17 cm in width and 35 cm in length. An impedance matching circuit maximizing the power transfer to the plasma load in this RF ion source, while taking account of technical feasibility, has been designed. For the impedance matcher design, estimation of plasma load impedances depending on operating conditions is of paramount importance because it determines the required impedance tuning range, voltage withstand capability, and cooling power. The electromagnetic model and hydrogen plasma global model were employed to predict the RF power absorbed by the plasma and plasma parameters (i.e., electron density and electron temperature) versus RF input power, respectively. Especially, in the case of rounded rectangular geometry compared to the circular cross section, the electromagnetic model demands for a numerical approach due to more complicated boundary condition. These models enable one to obtain the expected plasma impedances dependent on RF input power up to 50 kW, leading to the calculation of designed values about tuning capacitances of vacuum variable capacitors in the impedance matching circuit, voltages and currents of the antenna and capacitors at matching conditions. In this work, the analyses for the impedance matching circuit design and designed values as their results are presented and discussed in detail.
Requirements and modelling of fast particle injection in RFX-mod tokamak plasmas

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The planned upgrade of the RFX-mod device is a good opportunity to widen the operational space of the machine, in both RFP and tokamak configurations. Installation of a power neutral beam injector (NBI) is also envisaged and a NBI system compatible with RFX-mod is already available on site. It was previously installed in TPE-RX (Tsukuba, Japan), it has a nominal power of 1.25 MW, a nominal current of 50A and it can operate at a voltage of 25kV for 30ms (or 15 kV for 100ms). A porthole in the modified vacuum toroidal support structure is planned to be adapted to host this injector, which would operate in the equatorial plane and in a perpendicular direction with respect to the plasma column. This geometry is the only one possible due to mechanical constraints, mainly linked to toroidal field coils configuration. In this work, the METIS simulator is used to study the feasibility of TPE-RX injector integration in RFX-mod tokamak plasmas. METIS code allows the simulation of a full tokamak discharge in a time of the order of a few minutes by using relations coming from scaling laws coupled with simplified source models. The neutral beam injection in METIS is described by a decay equation applied in a simplified geometry and an analytical solution of the Fokker-Planck equation. In order to estimate a set of requirements for an NBI system to be applied to RFX-mod tokamak plasmas the beam shine-through has been carefully considered for a wide range of scenarios. Finally some indications of the physical effects to be expected from NBI-plasma coupling will be highlighted, with special attention to the beam energy absorption (both in terms of time dependent quantities and of spatial profiles) and to the possibility of inducing a transition from L to H confinement modes.
Neutral Beam injection has some well-established effects on plasma behaviour, such as the power threshold observed in L to H confinement mode transitions or the fast ion excitation of Alfvén modes, whose underlying mechanisms are still under investigation. In recent TJ-II experimental campaigns emphasis has been made in the characterisation of those Neutral Beam related effects. A study of Alfvén mode excitation as a function of beam parameters is under way, in which the dependence of the observed eigenmodes on the beam energy and current is examined. In order to carry out these studies, neutral beam parameters (beam energy and current) are made to vary in a wide range. To discriminate the effects of beam energy and beam current on plasma behaviour, beam energy scans are performed at a fixed current value and beam current scans at fixed energy. Such parameter scans involve beam perveance variations which affect the beam transmission, therefore a proper estimate of the power and fast particle current reaching the plasma calls for a previous determination of the beam transmission for each set of parameters. Beam transmission is determined by the beam divergence, which is usually obtained from the Gaussian beam power density profile at or near the beam focus. At TJ-II, beam profiles near the beam focus can be obtained by infrared thermography of the Target Calorimeter, a retractable target made of textured graphite (CFC graphite CX-1001U) that fully intercepts the beam at the duct exit [1]. After applying the appropriate coordinate transformation, beam divergence is obtained for each set of parameters in the energy and current scans. Computer simulation of the beam transport through the different apertures along the beam path finally yields the power and fast particle current transmitted to the plasma.
The latest progress of the 1st 5MW-NBI beamline on HL-2M

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The condition of 1MW-NBI heating for toroidal experiments to increase plasma energy storage and help making H-mode discharge had been well examined on HL-2A tokomak. A new tokomak with larger size and higher parameters named HL-2M tokomak which is under construction in Southwestern Institute of Physics of China needs higher auxiliary heating power, so a new NBI beamline with maximum 5MW injection power is designed. Since the Mid-2011 to the end of 2015, after a large number of numerical simulation, structure optimization and final factory manufacturing, most of the components including vacuum chamber, deflecting magnet, calorimeter, ion dumps etc and some sub-system have already been processed and assembled. Compared with the 1MW-NB injector on HL-2A tokomak, the size of 5MW-NB injector significantly increased. The specific performance is as follows: The total weight is nearly 50 ton, the total height is more than 11m, and the vacuum volume is around 20m$^3$. The 5MW-NB injector employs four positive bucket arc-driven sources with designed discharge parameters of filament current 2100A and arc current 1000A and could, in principle, inject 5MW D beam at 80keV particle energy or 3MW H beam at 60keV particle energy in 5 seconds pulse duration. Refer to vacuum pumping system, a cryopump system with pumping speed more than $1.4\times10^6$L/s is designed for the injector to maintain high vacuum degree. In order to test and improve ion sources, a ion source test bed was built and some experimental results were got. Presently the H ion beam power has been up to 66kV×26A and D ion beam power up to 66kV×20A.
H- extraction for development of a cesium free negative ion source using sheet plasma

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Production of negative ions plays an essential role in Neutral Beam Injection (NBI). A negative ion beam with an energy of 1 MeV and a current of 40 A (a current density of 20 mA/cm$^2$) is required for 3600 s to produce 16.5 MW of power. NBI predominantly uses negative hydrogen ion sources based on surface production. These negative hydrogen ion sources require cesium seeding to achieve a high ion density. However, cesium seeded surface-production of negative hydrogen ions is not desirable from the point of view of operating steady state ion sources. We demonstrated the production of negative hydrogen ions in cesium-free discharge by using the magnetized sheet plasma. We were confirm that there is negative hydrogen ions by mass spectrometry and performed an experiment of negative hydrogen ion extraction. Under a secondary hydrogen gas entering the hydrogen plasma, the peak position of the hydrogen plasma $n_{H^-}$ is localized in the periphery of the sheet plasma. It is found that hydrogen negative ions are formed by the dissociative attachment of low energy electrons ($Te = 1-2$ eV) to highly vibrationally excited molecules, which are attributed to the electron-impact excitation of molecules by high energy electrons ($Te > 10$ eV) in the plasma column, and are transported to the periphery of the sheet plasma. The hydrogen negative an ions density were detected using an omegatron mass analyzer, while the electron density and temperature were measured using a Langmuir probe. The maximum negative hydrogen ion beam is successfully extracted using grids located in the periphery of the sheet plasma. The negative hydrogen ion current density is about 20 mA/cm$^2$ at extraction voltage is 2 kV at a neutral gas pressure of 0.3 Pa and discharge current of 50 A.
Response of H- ions to extraction field in a negative hydrogen ion source

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In order to investigated the dynamics of H- ions and understand the extraction process inside filament-arc-driven plasmas in a Cs-seeded negative ion source, diagnostic experiments using a directional Langmuir probe combined with photodetachment measurement have been conducted. Two-dimensional flow pattern of H- ions has been obtained as well as the profile of H- ion temperature. The flow result shows that H- ions are produced on the surface of the plasma grid, initially launched backward to the beam direction and travel in the beam extraction region. H- ion energy decreases in the travelling and thermalized processes. During beam extraction, H- ion density decreases in the extraction region and the charge density is partially replaced with electrons. The decrement peak of the H- ion density is located at about 18 mm apart from the plasma grid. Comparing two-dimensional distribution of the H- flows before and during beam extraction, a motion map of H- ion responding to the extraction field has been obtained. It was found that the region corresponding to the decrement peak of H- ion density acted as an source of extracted H- ions. The results suggest that the surface produced H- ions are not extracted directly from the plasma grid but from the plasma volume in the Cs-seeded negative ion source using filament-arc discharge.
Design of high power RF amplifier for 3 MW/CW transmission line test rig

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India is developing 2.5 MW RF source at VSWR 2:1 in the frequency range 35-65 MHz for ITER project. Eight such RF sources will generate total 20MW of RF power for plasma heating and current drive. A large number of high power transmission line components are required for connecting various stages of RF source. To test these passive transmission line components at high power, a 3MW test facility based on the concept of Traveling Wave Resonator (TWR) is underway. For feeding power to the TWR using 10dB coupler, a high power (around 150 kW) CW RF amplifier is required. A tetrode based amplifier is designed to operate in class B operation, considering the optimized efficiency, gain and harmonics level. Grounded grid configuration is chosen for the stable operation and simplicity in mechanical assembly. The expected gain is around 14-15dB, therefore, 10kW Solid State Power Amplifier (SSPA) is chosen to drive this amplifier. The amplifier consists of input and output tunable coaxial cavities to cover the required frequency range along with support structure, tuning mechanism, transmission lines, high voltage and auxiliary power supplies and active cooling arrangements. High frequency electromagnetic simulation software Microwave Studio (MWS) is used for simulation and analysis of coaxial input and output cavities. Thermal analysis is conducted to check the adequacy of the amplifier design for continuous mode operation. The paper describes the detail design aspects, operational parameters derived from the data sheet of selected tetrode, simulation results, power supply requirements and thermal management for CW operation.
P4.035

Development of wideband solid state power amplifier for ICH & CD RF source

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ITER-India is developing Ion Cyclotron Heating & Current Drive (ICH&CD) RF source in the frequency of 35 to 65 MHz. Three cascaded amplifiers along with low power RF section, AC/DC power supplies and controls will be used for getting MW level RF power from one source. In the present configuration, two tube based tuned amplifiers, i.e. driver (~150 kW) and final (1.7MW) stage amplifiers are driven by a wideband solid state power amplifier (~ 10 kW). Development of such solid state power amplifier (SSPA) with required ±1.5 dB gain flatness in the above frequency range is very challenging, due to unique design of combiner and output matching circuit. This development is also aiming for achieving compact modular design, higher efficiency, usage of low voltage power supplies and better MTBF value compared with tube based amplifier of similar specification. Since 10 kW is needed as input power to the driver stage amplifier, the design goal for SSPA is to achieve power level of around 12 kW/CW. Considering losses in the combiner, total 16 pallet amplifier modules, each having capability of 1kW are to be combined using 16X1 wideband combiners. Each pallet amplifier module is designed using LDMOS transistors (MRFE6VP61K25H), which is capable to deliver 1250 W CW power in the required frequency range with adequate tune matching circuits. For input matching 9:1 ferrite based balun is used. For output circuit, 1:9 impedance transformation & balance to unbalance quarter wave transformer is used. For gate and drain supply voltages, adequate filters are designed and installed. In this paper, detail design and development of single pallet amplifier module as a part of wide band Solid State Power Amplifier will be discussed along with test results. Further, upcoming plan for integration of 16 such pallet modules with controls and monitoring system will be discussed.
Commissioning and first results of the reinstated JET ICRF ILA antenna

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The JET ICRF ITER-like Antenna (ILA) is composed of four resonant double loops (RDLs) arranged in a 2 toroidal by 2 poloidal array. Each RDL consists of two poloidally adjacent straps fed through in-vessel matching capacitors from a common Vacuum Transmission Line. Two toroidally adjacent RDLs are fed through a 3dB combiner-splitter. The JET ILA antenna has been operating at 33, 42 and 47MHz in 2008-2009 and has stopped operation in 2009 due to a failure of one of the tuning capacitors inside the antenna. Tests on a spare capacitor showed that a micro-leak was caused by the cycle wear of a capacitor’s internal bellows. The ILA was reinstated with a new operating scheme minimizing the full stroke requests of the capacitor. This contribution gives an overview of the works undertaken to reinstate the JET ILA up to the first results on plasma. The capacitors were replaced and high voltage tests of the capacitors were performed. An extensive calibration of all the measurements in the RF circuit was carried out. New simulation tools were created and control algorithms were implemented for the – toroidal and poloidal – phase control of the array as well as for the matching of the second stage. New protections are being implemented for the thermal and voltage protection of the capacitors. Low voltage matching tests were performed before the high power commissioning. Finally the first results on plasma are presented, showing that the new controls allow extending the range of the operation to lower (29MHz) and higher (49MHz) frequencies than previously achieved.
ITER-like antenna for JET first results of the advanced matching control algorithms

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The ITER-like Antenna (ILA) [1] for JET is a 2 toroidal by 2 poloidal array of Resonant Double Loops (RDL). It features in-vessel matching capacitors feeding RF current straps in Conjugate-T (CT) manner, a low impedance quarter-wave impedance transformer and a service stub allowing hydraulic actuator and water cooling services to reach the aforementioned capacitors. A 2nd stage phaseshifter / stub matching circuit allows to match the chosen conjugate-T working impedance to 30Ω. Toroidally adjacent RDLs are fed from a 3dB hybrid splitter. The assessment of the ILA results (2008-2009) identified that achieving routine full array operation required a better understanding of the RF circuit, tighter calibrations of RF measurements and last but not least a feedback control algorithm for the 2nd stage matching. The matching and phasing of the array is controlled by 22 feedback loops actuating the 8 matching capacitors, the 4 second stage phase shifters and 4 stubs, the 4 Main Transmission Line (MTL) phase shifters and the 2 phases with respect to a reference of the generators feeding the upper and lower half array through 3dB hybrid combiner-splitters. The circuit was extensively simulated [2] allowing to develop an algorithm which drives the 2nd stage matching circuit components, phase shifter and stub, to optimal locations with respect to the measured remaining VSWR excursions due to ELMs. The paper focusses on the new additional matching algorithms and assesses their performance. [1] F. Durodié et al., Plasma Physics and Controlled Fusion 54, 074012+ (2012), ISSN 0741-3335, URL http://dx.doi.org/10.1088/0741-3335/54/7/074012. [2] F. Durodié et al., AIP Conf. Proc. 1689, 070013 (2015); http://dx.doi.org/10.1063/1.4936520 *See the Appendix of F. Romanelli et al., Proceedings of the 25th IAEA Fusion Energy Conference 2014, Saint Petersburg, Russia
Neutral beam injection system for the C-2U field reversed configuration experiment

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In the C-2 field-reversed configuration (FRC) experiment, tangential neutral beam injection (NBI), coupled with electrically-biased plasma guns at the plasma ends and advanced surface conditioning, led to dramatic reductions in turbulence-driven losses. Under such conditions, highly reproducible, macroscopically stable, hot FRCs with a significant fast-ion population, total plasma temperature of ~1 keV and record lifetimes were achieved. To further improve the FRC sustainment and provide a better coupling with beams, the C-2 device has been upgraded with a new NBI system, which can deliver up to a total of 10 MW of hydrogen beam power (15 keV, 8 ms pulse), by far the largest ever used in compact toroid plasma experiments. The NBI system consists of six highly reliable and robust positive-ion based injectors featuring flexible, modular design. This presentation provides a comprehensive overview of the C-2U NBI system, including: 1) NBI test facility, beam characterization, and acceptance tests, 2) integration with the machine and operating experience, 3) improvements in plasma performance with increased beam power.

P4.039

Inverse heat flux evaluation from thermographic measurements in SPIDER

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To study and optimize negative ion production, the SPIDER prototype (beam energy 100 keV, current 48 A) is under construction in Padova, Italy. 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. As the front observation is not convenient, it is necessary to solve an inverse non-linear problem to determine the energy flux profile impinging on the calorimeter, starting from the 2D temperature pattern measured on the rear side of the tiles. The aim of the paper is to give an overview about different techniques which may be used to retrieve the flux profile, dealing with non-linearity and focusing in particular on the image filtering, which have demonstrated to be a very significant parameter.
Optimization of ITER poloidal field coil currents at initial magnetization phase

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An elaborate control of waveforms of poloidal field (PF) coils is prerequisite to ensure a reliable plasma start-up in ITER. An additional requirement in the ITER PF coil scenario development is that coil currents should be optimized to minimize quench risks during a discharge. In this paper, we use the quadratic programming method to optimize ITER PF coil currents at the initial magnetization (IM) state. We set the total magnetic energy of the system as a cost function to be minimized and calculate coil currents satisfying the initial magnetization flux and field null constraints compatible with ITER scenarios. The maximum initial magnetization flux is determined by trade-off between temperature margin and the magnitude of coil current at IM state. On the basis of this parametric study, we propose an optimized constraints and corresponding coil currents of initial ITER PF coil currents. All the optimization package is developed using the ITER Integrated Modeling Analysis Suite (IMAS) data structure for an easy adaptation to the integrated ITER simulator in the future.
Neural network implementation for ITER neutron emissivity profile recognition

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The ITER Radial Neutron Camera (RNC) is a diagnostic with multiple collimated inputs aiming at characterizing the neutron source. The RNC plays a primary role in the advanced control measurements and physics studies of ITER, and acts as backup for system machine protection and basic control measurements. The RNC primary design driver is the measurement of the neutron emissivity radial profile within specified measurement requirements regarding temporal and spatial resolution and fusion power. This paper presents a method based on neural network methods to provide an estimate of the neutron emissivity profile in different deuterium-tritium ITER scenarios and for different RNC architectural configurations which are under investigation [1]. The design and optimization of the feed-forward neural network with back-propagation algorithm and the choice of the training data sets will be discussed. The effect of statistical noise and background are included in the neural network supervised learning phase. A decision algorithm has been implemented to select which inverted neutron emissivity profile gives the best estimate of the real one. The profile recognition based on neural networks is sufficiently fast that it is considered feasible for a real time environment [2]. This study indicates that neural networks can achieve an accuracy and precision within the spatial and temporal requirements set by ITER. The following aspects of the neural network implementation will be discussed: i) the decision algorithm requirements of a priori knowledge of the plasma flux surfaces; ii) the role of ensemble averaging of multiple static predictors and iii) the effect of missing data. [1] D. Marocco et al., System Level Design and Performances of the ITER Radial Neutron Camera, IAEA 2016 [2] N. Cruz et al., The Real-Time Software Design for the ITER Radial Neutron Camera, this conference.
P4.042

Preliminary Exception Handling Analysis for the ITER Plasma Control System

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To operate ITER and control long and finally thermonuclear discharges with very complex physics and a limited set of actuators requires a sophisticated Plasma Control System (PCS). To provide the required control functionality, the PCS will include many control loops to keep parameters within operation envelopes. These must be backed by exception handling functions, to optimize continuous control performance, autonomously perform controlled shutdown in critical situations, or assist interlock systems in pulse termination for investment protection. The Conceptual Design Review (CDR) for ITER PCS held 2013 covered all ITER operation phases. It provided a comprehensive set of system requirements and a preliminary functional analysis, and proposed a conceptual PCS design compatible to these. To prepare for the Preliminary Design Review (PDR) in late 2016, the PCS analysis and design is currently being extended and detailed. The goal is to show that candidate designs meet ITER requirements and assumptions, with focus on control of 1st plasma and early H/He phases of ITER operation (but keeping future high-performance phases in mind). Further aspects to be covered are integration of PCS exception handling with plasma state forecasting and with ITER CIS (Central Interlock System) protection mechanisms. Capturing and analysing the required exception handling capability follows a formal approach (see Treutterer et al., this conference): Separate research groups collect functional aspects needed to control (and handle exceptions during) nominal discharge phases (magnetic and kinetic control of plasma initiation, formation, ramp-up, flattop, ramp-down), or when disruptions and runaways occur (includes interfacing to CIS). In this contribution we show how such results enter a database and how this information is used to find out how to organize exception handling, find recurring patterns, synthesise re-usable handling schemes, develop standardised methods to categorize and escalate exceptions and manage a rule-based decision taking throughout the system.
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Design of the ITER plasma control system is proceeding towards its next - preliminary design - stage. During the conceptual design in 2013 an overall assessment of high-level control tasks and their relationships has been conducted. The goal of the preliminary design is to show, that a reasonable implementation of the proposed concepts exists which fulfills the high-level requirements and is suitable for realistic use cases. This verification is conducted with focus on the concrete use cases of early operation and first plasma, since these phases are mandatory for ITER startup. In particular, detailed control requirements and functions for first plasma operation including breakdown, burn-through and ramp-up in L-mode, as well as for planned or exceptional shutdown are identified. Control functions related to those operational phases and the underlying control system architecture are modeled covering nominal operation, as well as critical or frequent exceptional cases requiring off-normal response. The goal is to check whether the flexibility of the conceptual architectural approach is adequate also in consideration of the more elaborate definitions for control functions and their interactions. In addition, architecture shall already be prepared for extension to H-mode operation and burn-control, even if the related control functions are only roughly defined at the moment. As a consequence, the architectural design is amended where necessary and converted into base components and infrastructure services allowing to deploy control and exception handling algorithms for the concrete first-plasma operation. The investigations will be backed up with simulations comprising simplified plasma models as well as control functions embedded in the envisaged control system architecture.
3D analysis of magnetic lines variations at breakdown due to error fields in ITER

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The magnet system in ITER is composed by three main coils groups, characterized by tight tolerances on manufacturing and assembly, to keep error fields at levels compatible with plasma operation. Additional coils correct error fields guaranteeing suitable accuracy at start of flat top [1]. Plasma initiation in ITER will be critical, since low electric field will be available, and a reduction of field lines connection length may occur due to stray fields [2]. A number of studies have been performed to assess impact of different stray field sources, but they mostly used equivalent 2D models to assess effects of intrinsically 3D fields [3], or were performed on other machines run in “ITER-relevant” modes. Magnet tolerances will provide local contributions to stray field at breakdown in the order of some mT, whose effects must be assessed using 3D models. The toroidally averaged figures are much smaller (in the order of fractions of mT), but impact in terms of e.g. connection length of localized field map deformations may provide deeper understanding of field at breakdown. In this paper, starting from a waveform scenario optimized using a axi-symmetric model for nominal coils and vessel, a full 3D numerical model is used to analyse the contribution to magnetic field in the breakdown region due to tolerances on coils, including toroidal field coils. Advantage is taken from semi-analytical parallel computing routines for high accuracy field computation, and two fast, 3D line-tracing algorithms are used to assess average stray fields and connection length reduction. [1] ITER EFDA Documentation Series No.24, “ITER Technical Basis”, Chapter 3.7.4.1.3 [2] Gribov Y. et al., “Chapter 8: Plasma operation and control”, Progress in the ITER Physics Basis, Nuclear Fusion 47 (2007), S385. [3] A. B. Mineev et al., “Study of ITER First Plasma Initiation using a 3D Electromagnetic Model”, FEC 2014.
Comparison of three methods for the solution of eddy current problems in fusion devices

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We compare three methods for the solution of eddy current problems arising in fusion technology. We first consider the Finite Element Method formulation based on the reduced magnetic vector potential [1]. This formulation provides a very sparse system matrix and is able to solve problems on meshes composed of tens of millions elements. Yet, it requires to produce the mesh for both conducting and insulating regions, something which is very time consuming. That is why as a second method we consider the volume integral formulation in term of the electric vector potential [2], which requires to mesh only passive conductive structures but having the drawback of dealing with a full system matrix. The third method is based on iteratively solving a Poisson problem in the conductive region and computing a correction magnetic field with the Biot-Savart law [3]. The Poisson problem, solved with a formulation providing a solenoidal current density [4], gives an estimate of the current density by enforcing the Faraday’s law. Then, an updated and solenoidal magnetic induction field is obtained form the computed current density with the Biot-Savart law. This magnetic induction field will be in turn used as a source in the Poisson problem. The iterations stop when the computed update is negligible, which means that a self-consistent solution of Faraday’s and magnetic Gauss’ laws if found. Pros and cons of the proposed methods are assessed on a benchmark problem, i.e. eddy currents induced in ITER-like 3D conducting structures. [1] O. Biro et al, IEEE Trans Magn 25 (1989) 3145-3159 [2] R. Albanese et al, IEE Proc. A 135 (1988) 457-462 [3] T. Takagi et al, IEEE Trans Magn 24 (1988) 2682-2684 [4] P. Bettini et al, JCP 273 (2014) 100-117
Interface challenges as part of the ITER plasma control system design

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The ITER Plasma Control System (PCS) is now approaching the second phase of development, the Preliminary Design Review (PDR). The PDR, expected at the end of 2016, is now more deeply investigating possible solutions for the different control areas aimed at operations up to 15MA with low auxiliary heating in L-mode. The entire sequence of a plasma discharge from the break-down to the termination, including disruption, has been addressed by the ITER PCS team and a group of experts. The main challenges have been considered and the first proposal for controls has been delivered. The control functions in the PCS are strictly linked to the performance of the ITER actuators and diagnostics. The capabilities of those systems need to be carefully validated against the control needs. System and performance requirements shall be consistent with the control schemas and where limitations or restrictions are identified, it is necessary to provide alternative solutions. To guarantee this consistency, a set of interface documents is being prepared. Those interfaces for each of the plant systems that can impact the PCS activities detail the requirements specifically needed for control and report also the functional relationship between the two systems. PCS has also to consider areas not actively part of plasma control that might affect or limit PCS operations (i.e. forces in the superconducting coils). This paper reports the main outcome from the interfaces definition. The actuator boundaries and plant systems constraints impacting the PCS design will be presented. For the sensors the challenge is the derivation of real-time measurement requirements in relation to the separate diagnostic requirements and their respective interface with PCS. The complex organization of data integration with the PCS will be discussed. The paper will present the status of the development in these areas and the planned path of the PCS development.
Experiments on actuator management and integrated control at ASDEX Upgrade

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Integrated control of many plasma parameters simultaneously is expected to increase the reproducibility and stability of scenarios, which are otherwise developed laboriously through trial and error. The benefits are expected to be especially important for high performance scenarios, operating near multiple stability boundaries. The two main challenges of integrated control are: firstly the physics coupling between parameters, which is often non-linear and regime-dependent. Secondly, fusion experiments operate with a limited set of actuators which restricts the set of parameters that can be simultaneously controlled. Furthermore, one actuator can affect several parameters and/or several actuators can affect the same parameter. To satisfy conflicting requirements on actuators, some form of actuator management is required. For the specific case of actuator management for ECRH, a routine has been developed at ASDEX Upgrade which optimally allocates four gyrotrons to central heating (to avoid tungsten accumulation), mitigation of 3/2 Neoclassical Tearing Modes (NTMs) and mitigation of 2/1 NTMs. The allocation occurs in real-time, reacting to changing plasma conditions and actuator availability during an experiment. In future this could be extended to more gyrotrons and targets such as sawtooth control, electron temperature profile control and current profile control. Experiments will show the flexibility of this algorithm to operate in a range of scenarios and in combination with other controllers for beta, radiation, edge density using gas puffing and core density using a new pellet feedback controller. The controllers operate in parallel, and hence integration will be limited to co-ordination of their reference trajectories by a high level supervisory controller. This contribution will present observations on the coupling between controllers, and a design for a truly integrated MIMO controller to be developed for the next MST1 campaign.
Real-Time reflectometry - an ASDEX Upgrade DCS plugin-App for plasma position and shape control

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On future long pulse fusion devices an extended set of diagnostics will play an increasingly important role in advanced plasma control. In particular, O-mode microwave reflectometry will be used, on ITER and foreseeably on DEMO, to complement the standard magnetic diagnostics for plasma position control. With the preliminary design of ITER’s plasma position reflectometers (PPR) presently underway, it is of the utmost interest to test beforehand all possible aspects of this future control application. ASDEX Upgrade is the best suited experimental facility on which such tests can be performed. Not only it features a modern, modular and easy adaptable control system, but also the only O-mode reflectometry setup capable of performing simultaneously measurements on two of the four lines of sight of ITER’s PPR. After the first successful demonstration of plasma position control using Low-field side (LFS) reflectometry density profile data, the real-time (RT) diagnostic’s hardware was updated to acquire a higher number of signals and to improve its RT data-processing capabilities. Meanwhile, the system’s software was rewritten to implement a pipelined architecture that improves the deterministic behavior of the diagnostic’s internal data flow, from data acquisition up to control data delivery to the AUG discharge control system (DCS). The pipeline last stage, that calculates the control relevant parameters, synchronizing and communicating with the DCS, now uses the new DCS software framework, appearing to the DCS infrastructure as a modular plugin RT diagnostic “App Process”. Herein are discussed the performance and reliability gains obtained with this new software implementation. Fault tolerance, measurement rate and implementable synchronization strategies allowed by the new DCS infrastructure are also addressed. Experimental data from discharges where reflectometry is used for controlling both plasma position and shape are presented to assess the system “real life” performance.
Development of the control algorithm for advanced divertor configuration

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The advanced configurations (snowflake and tripod) have been designed with EFIT based on current poloidal field (PF) coils system of HL-2M to study the advanced divertor physics and support the high performance plasma operation. The characteristic parameters of the advanced configuration (the distance between two X-points, magnetic flux expansion and weak field area and so on), especially the position of the second X-point, are the key factor for the advanced divertor studies. So the active magnetic control (especially X-points control) of the advanced configuration is essential for the operation and the study on physical mechanism of the advanced divertor. A control algorithm for the advanced divertor configuration has been developed. The relationship between the X-point position and the PF coils current has been created by this algorithm with the magnetic filed of four control points near the divertor area. These magnetic filed can obtained from the RTEFIT. In addition, the isoflux control and the influence of the plasma motion have been also considered in the control algorithm. With this algorithm, the characteristic parameters, especially the position of the X-points, can be accurately tuned by regulating the current in PF coils (especially in divertor coils), meanwhile remain the plasma main parameters. For example, the second X-point (the first X-point is located at the separatrix of main plasma) can fluctuate around target plate of divertor, and the distance between two X-points can vary from 25cm to more than 50cm. These results can be useful for deeper understanding of the advanced divertor operation, and providing the reference database for determining whether the advanced divertor as a geometrical solution to reduce heat loads on the divertor plates for ITER or fusion reactors based on tokamak.
Laser-induced Fluorescence for ITER Divertor Plasma

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Laser-induced fluorescence (LIF) diagnostic system on ITER will be used for local measurements of helium density (nHe) and ion temperature (Ti) in the divertor region. The diagnostics is combined with divertor Thomson scattering (DTS) via common laser injection and signal collection optics. Physical aspects of the LIF method for measuring the plasma parameters and the layout of the system elements are in the focus of this work. The LIF method is based on laser pumping of the allowed transition between the excited states of the atom / ion and subsequent registration of the fluorescent radiation of the same or other transition. Dye lasers, optical parametric oscillators (OPO), Ti:Sapphire lasers tunable in the visible and near UV ranges can be used as the pulsed excitation sources. A Nd:YAG laser with an OPO is preferred for the density measurements (nHe) due to simplicity of its design, broad spectral line and wide operating range (380-1100 nm). Dye lasers allow both extra narrow (3-5 pm) and wide (up to 300 pm) spectrum generation. They can be used in the applications requiring continuous tuning of the laser wavelength (measurements of Ti). A new approach for Ti measurement at one time-point using two synchronized tunable dye lasers will be presented. Multichannel filter polychromators similar to those of DTS will be used to observe the signals on the interesting fluorescent lines. The active LIF signals and background radiation are estimated using the developed collisional-radiative models. The estimations demonstrate the ability to measure nHe with temporal resolution 20 ms and the errors $\Delta$ nHe < 20\% in both He and DT-regimes. Ion temperature can be obtained from the Doppler broadening. The accuracy of different approaches for the Ti measurements will be estimated.
Testing of uncooled sandwich molybdenum mirrors for H-alpha and visible spectroscopy in ITER

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First mirrors (FMs) for ITER optical diagnostics induce a number of specific requirements including low sputtering rate, high neutron/gamma radiation and thermal stability to keep the optical performance in the DT plasma shots. Additionally, the FM surface must withstand the discharges by a cleaning system aimed to eliminate Be deposits. A number of experiments have shown that the mirrors made of single crystal molybdenum (SC-Mo) retain their reflectivity under surface sputtering by high energy particles in plasma devices. Unfortunately, large SC-Mo ingots for manufacturing of the full size mirrors for ITER from a single piece are not commercially available. A sandwich-like design has been proposed in which the mirror is made of a few SC-Mo segments bonded to the polycrystalline molybdenum (PC-Mo) substrate. Since the FMs for H-alpha and Visible Spectroscopy (HA&VS) is planned to be uncooled, their temperature can reach 350-400°C, and a difference in CTE of SC-Mo and PC-Mo may affect the thermal stability of the sandwich structure. In order to test the thermal stability of the sandwich SC/PC-Mo mirrors, two full-scale prototypes of the flat and the focusing FMs were manufactured according to the current HA&VS design. The 200×60×40 mm samples were made of two 200×30×4 mm SC-Mo segments bonded to the PC-Mo plate by the hot isostatic pressing technology. Few cycles of thermal heating in vacuum have been performed up to 350°C. The curvature of their surface during testing was monitored optically by illuminating the mirrors with a set of parallel beams and analyzing the variation of the directions of reflected beams. As a result, minor changes in the mirrors’ curvature \(\sim 10^{-3}\) m were observed in a heated state. After cooling down the mirrors’ shape return back with high accuracy. The results prove the applicability of current approach to the uncooled FM design for HA&VS.
Fiber optics for plasma diagnostics in ITER

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Silica-based optical fibers have a high light transmission in visible range and so they are widely used for transmitting the light from plasma to detectors in modern thermonuclear facilities. The fiber bundle is comprised as a rule of several tens or hundreds optical fibres of 100-500 microns diameter. The lifetime of the optical fiber in ITER should be more than 15 years. Radiation resistance is a major problem for fiber optics in ITER because of the light transmission has to be adequate to allow measurements up to the end of DT mode. The report is devoted to problems of the development of fiber optics for ITER diagnostics. Options for fiber bundle are considered on the example of H-alpha diagnostics. Fiber bundles start in the port cell held there for about 6 m to a concrete wall and then extend up to the diagnostic room at a distance about 130 m. The results of neutron fluxes calculation at the location of the fiber bundles are given in the report. A brief review is presented of the irradiation tests of optical fibers. The opportunities for optical fiber radiation hardening or annealing are discussed. It is shown that the pure silica-core/F-doped silica-clad fibers are the best from point of view of radiation resistance in ITER. Of course, these fibers are not perfect, because they suffer from radiation-induced optical absorption and luminescence. To improve the behavior of optical fibers under irradiation they must be in the place already well protected from neutron flux. Analysis of the irradiation tests and neutron calculations shows that the radiation resistance of modern optical fibers provides the opportunity for its application in ITER in the spectral range 450-1300 nm.
Optimization of optical dumps for H-alpha spectroscopy in ITER

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The performance of ITER Main Chamber H-alpha & Visible Spectroscopy is challenged by the problem of separating the contribution of visible light emitted in the scrape-off-layer (SOL) from the background of much higher intensity, produced by the divertor stray light (DSL) reflected by the all-metal first wall (S.Kajita, et al., PPCF, 2013). A differential (bifurcated-line-of-sight) measurement scheme was proposed to solve this problem (24th IAEA FEC, 2012, ITR/P5-44). It assumes simultaneous observation of two neighboring fields on the first wall with noticeable local difference in the wall reflection Rw, e.g. ensured by using the optical dumps (OD) with reduced reflectivity of incident light. However, the differential scheme could be effective, provided the spectral intensity of SOL emission and the normalized line shape of DSL spectrum are not perturbed by the OD. Thus, the proper OD design should minimize the perturbation of DSL line shape, caused by 1) spectral non-uniformity of DSL directivity at the OD entrance pupil, and 2) the difference of bidirectional reflectance distribution functions (BRDF) of the OD and the first wall. The effect of the OD design on the BRDF and the accuracy of the SOL emission measurements are evaluated for the few lines of sight directed from ITER equatorial port to the ODs located in blanket modules ##1-8 by simulation with the use of Zemax OpticStudio software. The data of predictive modeling of the flat-top stage of inductive mode of ITER operation, derived by the SOLPS4.3 (B2-EIRENE) code are used to calculate the spectral and angular profiles of the incident light on the ODs. A number of OD models are considered, and the respective BRDFs and measurement errors are evaluated for the certain ITER plasma discharge scenario (low-density far-SOL L-mode) in order to optimize the OD design.
P4.054

Investigations on camera integration and data acquisition architecture for the ITER equatorial Vis/IR diagnostic

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The ITER equatorial visible and infrared wide-angle viewing system is a first plasma diagnostic that will be used to image the visible plasma boundary and the in-vessel components temperatures for real-time machine protection and plasma control purposes, as well as offline physics studies. The system will be installed in four equatorial ports and will have 15 lines of sight covering most of the in-vessel component surface. The preliminary design of the system is under development by a consortium of academic and industrial partners on behalf of the European domestic agency (F4E). This project is currently at the system-level design stage where different design options including optical design, integration of key components and back-end electronics are elaborated and assessed in order to select the best adapted architecture. In this context, this paper reports on the current achievements on sensor and data acquisition hardware architecture options and on the physical integration of the cameras in the port-cell environment. The sensor pre-selection is based on a market survey of visible and infrared cameras. A scoring metric has been defined to compare the different camera models in terms of relevance with the current measurement requirements. A global configuration is then drawn for the complete system. The proposed data acquisition hardware architecture takes into account the long distance to transport the camera signal, the camera data throughput and the ITER recommendations for hardware selection. The investigation on the camera integration relies on the available volume in front of the optical line outputs in the port-cell, the nuclear radiation levels and the human access limitation for maintenance operation. The goal is to propose a concept of a shielded cabinet able to host the defined number of cameras, providing local shielding for the electronics and minimizing the human presence for camera maintenance activities in the port-cell area.
Load specification and vibration test of the ITER equatorial port H-alpha diagnostic dogleg mirror

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The H-alpha and Visible Spectroscopy Diagnostic shall measure spectrally, spatially and temporally resolved emission of hydrogen isotopes and impurities in the ITER scrape-off layer. There are four H-alpha diagnostic channels, located in 3 port plugs. In the current design status, all main interfaces have been iterated with the Port Integrator. All major subsystems, of this complete end to end design for spectral and imaging channels, are defined in detail and their functional compliance w.r.t. the system performance and structural integrity has been assessed by analysis. The thermal and mechanical loads to be sustained from the in-vessel components are high. This is the case for the First Mirror Unit (FMU) and hot dogleg (HDL) mirrors, making the labyrinth between the FMU and the vacuum window assembly in the Port Plug. The compliance of the HDL design w.r.t. the system performance and structural integrity has recently been successfully verified by vibration tests performed on a HDL breadboard. The full preparatory work starting with IO-CT delivered set of appropriate power spectral density (PSD) and followed by the test predictions with ANSYS, will be described in detail. The delivered PSD has been adapted to our nodes taking the appropriate floor response spectra into account. Moreover the displacements of the HDL M3 node as function of time has been used to compute the corresponding accelerations as input for the PSD calculation. Finally an envelope of all PSDs was constructed and used as input for the random vibration test for all axis of excitation. This presentation describes the test approach, procedure and successful results of the test campaign. It will be explained how vibration loads to be applied to the system have been derived. The analysed results will be presented together with images of the setup and videos of the vibration test.
Design & analysis process and implications on the optical performance

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The H-alpha and Visible Spectroscopy Diagnostic shall measure spectrally, spatially and temporally resolved emission of hydrogen isotopes and impurities in the ITER scrape-off layer. Four H-alpha diagnostic channels are designed to observe the plasma. They are located in 3 port plugs:

- Equatorial Port #11: TV (Top View): poloidal wide FoV covering the upper part of the inner wall BV (Bottom View): poloidal wide FoV covering the lower part of the inner wall
- Equatorial Port #12: a tangential wide FoV channel covering the outer wall
- Upper Port #02: a wide-FoV channel covering the divertor. The H alpha diagnostic in the equatorial port systems are scheduled for first plasma. In the current design status, all main interfaces have been iterated with the Port Integrator. All major subsystems of this complete end to end design for spectral and imaging channels are defined in detail and their functional compliance w.r.t. the system performance and structural integrity has been assessed by analyses. The first opto-mechanical design has been iterated between the optical and mechanical designer. The initial FEM analysis focusses among others on the determination of temperature profile, deformation of mirrors for system performance assessment and a first qualitative EM analysis. The output is then retrofitted into the mechanical model but also into the optical model to determine the impact on the optical performance. This step wise approach has been applied for the iteration between the different design and analysis disciplines such as optical design, mechanical design and structural, thermal and EM analysis on the other side. The design and analysis approach has recently been successfully verified by a vibration test of some of the components. The iterative analysis approach which has been followed, will be detailed here, highlighting the emphasis of each analysis phase.
The UWAVS R&D and design efforts for ITER

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One of the diagnostic systems being provided by the US is the Upper Wide Angle Viewing System (UWAVS), which provides real-time, simultaneous visible and infrared images of the ITER divertor region via optical systems located in five upper ports. The UWAVS is designed in three main sections: in-vessel, interspace and port cell assemblies. Each assembly utilizes multiple steering and relay mirrors to direct the in-vessel light out of the tokamak to the port cell camera sensors. The primary design challenge of the in-vessel assembly is maximizing performance of the overall system while surviving the severe electromagnetic and nuclear ITER environment. A first mirror material study was conducted and determined molybdenum was the best choice for the first two mirrors in the optical train. A fail open, bellows actuated shutter with cross pivot flexure design was determined the most reliable mechanism to protect the first mirror. A geometrically representative glow discharge mirror cleaning system is being designed and will be tested to maximize cleaning efficacy while minimizing optical degradation of the molybdenum mirrors adjacent to the plasma. The shutter and first mirror assemblies were packaged and designed for replacement via remote handling methods to minimize radiation waste and cost associated with these eventual component replacement. The preliminary optical and structural design provides a robust and reliable system while maximizing the field of view. Analysis results verify all optical and structural performance criteria are being met with positive safety margins. The R&D efforts, the technical challenges and issues, and the design and analysis results are presented.

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Hardware solutions for ITER divertor thomson scattering

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ITER Divertor Thomson scattering (DTS) was discussed in a number of presentations and papers. The development of diagnostic equipment for ITER DTS is under way and coming to its conclusion. Choice and justification of lasers and polychromator design as well as first mirror protection are the focus of the presentation. Q-switched Nd:YAG laser for DTS in ITER (1.064mm, 2J, 50Hz, 3ns) is implemented. To minimize the energy loss caused by divergence, the near field image must be transmitted to the first laser mirror requiring the uniform energy distribution across the laser footprint. Expecting that laser diode pumping instead of flash-lamp will improve the near-field energy distribution as well as considerably increase its service life, the diode-pumped version of the laser is under development. Compact and powerful DTS acquisition system (5 MS/s, 12 bit) is prepared to embed the digitizer into DTS spectrometers. The chief reasons for this are: fiber input with TS signal and fiber output with digitized signals protect from electromagnetic interferences; does not require additional cooling for low power electronics, providing more reliable operation and compact design than in previous polychromators; an order higher sample rate (5 MS/s instead of 500 MS/s) allows using oscilloscopic mode for nanosecond laser pulse measuring; considerably lower ADC cost saves the diagnostic budget. Good progress was achieved in the development of optics protection. Specifically shaped diagnostic channel was suggested to prevent beryllium deposition caused by hydrodynamic fluxes. This approach was verified on simulators of dynamic loads on ITER divertor - Magnum, Pilot and QSPA. RF plasma cleaning of metal deposits was thoroughly investigated. Common approach, which can be routinely used in ITER by all optical diagnostics is under development. The DTS on Globus-M at Ioffe Institute commissioning will provide demonstration and testing of the equipment performance and technical solutions developed for ITER DTS.
P4.059

Radiation shielding design evaluation for detectors of ITER VUV Edge Imaging Spectrometers

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The local shielding design for the detector of ITER VUV Edge Imaging Spectrometer is optimized based on the MCNP calculation using a local port cell model of ITER Upper Port #18. A back-illuminated CCD, the envisaged VUV detector for ITER VUV Edge Imaging Spectrometer will be installed at ITER Upper Port #18 port cell region, in which a harsh radiation environment is expected with neutron flux higher than $10^5$ neutrons·cm$^{-2}$·s$^{-1}$ mainly originated from 14 MeV d-t neutrons as well as high gamma ray dose of several tens kGy from the plasma, activated materials and $^{16}$N isotopes in water coolant. To evaluate radiation exposure to the detector, the local port cell model is developed as well as the boundary neutron and gamma ray source model based on MCNP result using C-Lite, which is expected to reduce both calculation time and statistical error. Since the radiation exposure to the back-illuminated CCD should be mitigated as much as possible to minimize the radiation damage to the detector as well as single event upset, local shielding design options for the VUV detector with various shapes, thicknesses, and material compositions are evaluated. In this paper, preliminary radiation assessment for ITER Upper Port #18 port cell region and the optimized shielding design for VUV detectors with estimated shielding factor will be presented.
ITER Core CXRS diagnostic: Assessment of different optical designs with respect to Neutronics criteria

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The Charge Exchange Recombination Spectroscopy (CXRS) diagnostic aims to measure emission lines of impurity isotopes in the ITER plasma in order to quantify several parameters like the composition of the plasma (density of helium, deuterium or tritium), the ion temperature or rotation velocities. The core plasma CXRS shall be installed in one of the ITER Upper Port Plugs (UPP #3). Currently, four different optical layouts are being assessed with respect to the optical performance, engineering feasibility, cost, maintenance especially with respect to remote handling and the performance of their neutron radiation shielding. This work is devoted to the neutronic analysis performed in support for the design of ITER CXRS-core Diagnostic System, presently under development by the IC3 Consortium (FZJ, KIT, BME, Wigner RCP, TU/Eindhoven, FOM-DIFFER, CCFE, CIEMAT, Optimal Optik). In the paper, results of the neutronic analyses are presented showing the differences between the four different designs with respect to several nuclear responses such as neutron and gamma fluxes, dose rates in the port interspace, maps of nuclear heating including the CXRS (focussing on critical components such as mirrors, shutter and window), the toroidal/poloidal field coils as well as the vacuum vessel. Furthermore, radiation damage maps were calculated covering large areas of the upper port plug and of its environment. The results indicate the viability of one of the preferred designs from the neutronic point of view but also show the potential for improvements.
JUVIL: A New Innovative Software Framework for Data Analysis of JET Imaging Systems

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Imaging systems are an indispensable technique for successful plasma operation of fusion devices. At the JET tokamak, numerous cameras in the VIS/NIR/MWIR spectral ranges are used for plasma physics studies as well as for the real time overheating protection of the first wall and for live plasma monitoring during operation. The protection system, on the basis of the NIR imaging cameras, is routinely used on JET for monitoring the surface temperature. A new powerful software framework JUVIL (JET Users Video Imaging Library) has been developed and successfully installed at JET for fast data visualization and advance analysis of all types of imaging data. The JUVIL framework is based on modular object-oriented components implemented in Python to simplify work with JET scientific data. It provides standard interfaces to access video data and post-processing, which are highly configurable and can be easily extended and adapted for new data formats and imaging cameras. One of the GUI components is the video player, widely used during the last JET campaign. It displays the video data for VIS/NIR/MWIR cameras and automatically carries out the post-processing (image rotation, data format conversion, scaling of non-interlaced fields to full frames). In addition, JUVIL loads the video background, calibration file, performs dead pixels and flat field corrections and provides general information such as frame geometry, camera filters, and exposure time. The software is able to plot maximum temperatures of ROIs (Region of Interest) from the Real-Time Processing System and to calculate maximum/average temperatures for the user’s own ROIs. There is an interface to retrieve VTM (Vessel Thermal Map) events, wall segments maximum temperatures and thresholds for each plasma pulse and to load the corresponding video image if a hot spot alarm is detected. Finally, the JUVIL hotspots editor is able to store the hotspots parameters and analyse their evolution.
Pre-Emptive Data Caching Infrastructure for Data Centric Analysis and Modelling

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The next generation of tokamaks, e.g. ITER, will have extremely large data collection rates (\textasciitilde0.3PBytes per day), significantly larger than those experienced today, with consequential new challenges in data management, data analysis and modelling. With long pulse durations it is important that data be accessible during the experiment for plant monitoring in quasi real-time analysis. One of the big data challenge for these use cases is to ensure that appropriate data is very quickly made available when it is required and where it is consumed. Data volumes with limited network capabilities mean not all data can be distributed in time – we have to be selective. How is this best achieved? One solution is to pre-emptively identify and distribute data to local cache before an application or model requests it. Pre-emption will rely on analysis of historical access patterns using data mining techniques to identify a set of rules whereby following an initial data request the most probable set of next requests can be inferred. Apache Spark is being used to capture the inference rules from IDAM data access logs accumulated from the MAST experiment over several years (>50 Million records). Implementation of these rules requires the inferred sets of data be copied to cache on the host computers running the application code ahead of the next data request. This work is part of the SAGE EU H2020 project, led by Seagate, developing exascale data centric computing architectures. The SAGE hardware consists of multi-tiered storage and HPC compute nodes where data are moved between tiers to where needed using the concept of percipience. The work presented will describe the Spark workflow, the results of the analysis, and an implementation of the pre-emptive caching infrastructure at MAST, together with plans for its implementation and testing on the SAGE platform.
Boosting learning for robust classification of TJ-II nuclear fusion databases

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Huge databases are a common situation in fusion. Physical properties of plasma are studied by thousands of signals, sampled at very high frequencies, producing enormous amount of data. A medium-size nuclear fusion device such as TJ-II can generate discharges that last around 500 milliseconds, reaching up to 100 Mbytes per one simple shot. Larger fusion devices such as JET can produce 10Gbytes per discharge, and ITER could storage 1Tbytes per a 30 minutes shot. The thousands acquired signals involve the analysis of data in high-dimensional spaces. In such spaces, the data become sparse, which makes difficult the searching of patterns with similar properties, reducing the efficiency and increasing the overfitting of learning algorithms. This issue, which is known in the literature as the course of dimensionality, can be faced by using suitable feature extraction methods to reduce the input space into a low-dimensional space. However the selection of feature reduction techniques is not straightforward, and commonly it is a time consuming task that requires an important effort. During last years the use of boosting algorithms is become very popular to avoid overfilling and to obtain generalized classifiers in problems with high-dimensional spaces. Boosting is an approach to machine learning to achieve a highly accurate and robust classification by combining many relatively weak and simple rules. The AdaBoost algorithm was the first practical boosting algorithm, and is one of the most widely studied, with applications in several fields. This article describes the use of AdaBoost for building robust classifiers of patterns in fusion databases. In order to show the benefits of the approach, images from the Thomson Scattering diagnostic, and time-domain signals of the TJ-II database have been tested. The work includes a comparative study with previous results of other classifiers built with support vector machines and artificial neural networks.
The "voiceprint" of a tokamak and its application in monitoring

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The noises of a tokamak during operations form the “voiceprint” of a tokamak. By installing a set of microphones in several optimized positions around the tokamak machine, most noises can be detected and can be used as the “voiceprint” of the tokamak for monitoring its status. Noises of a tokamak in discharge-ready status are mainly continuous and/or cyclical noises from pumping system, water cooling system, etc. We can define the noises of each system under normal operation status as a set of reference voiceprints and extract Mel frequency cepstral coefficient (MFCC) features from them. Then, if the real time MFCC spectrum of the noises detected differs from the reference MFCC spectrum, it means that the pumping system or something else is in an abnormal status. The voiceprint monitoring software will find and compare the difference with each MFCC features, give an alert to the operators, and specify the possible malfunctioning systems. The machine voiceprint monitor is easy to be deployed and doesn’t have any electrical or mechanical contacts with the existing machines. It is very convenient to use.
90 degrees cylindrical energy analyzer for the plasma potential fluctuations measurements on the tokamak ISTTOK

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The heavy ion beam diagnostic of the tokamak ISTTOK is operated with a 20 keV Xe\(^{++}\) ion beam and a multiple cell array detector to collect the secondary Xe\(^{2+}\) ions created along the primary beam path by ionizing collisions with plasma electrons. In this multichannel mode of operation, the use of standard Proca-Green 30\(^o\) parallel plate energy analyzer for the plasma potential measurements is of reduced applicability to collect all the secondary ions along the plasma cross-section. The multichannel time-of-flight method has been implemented in the recent past as an alternative approach. However, the results of the measurements have indicated a lack of enough resolution for investigations of plasma potential fluctuations in the \(\Delta E/E \sim 10^{-4}\) range. In this presentation a multi-channel multi-slit 90\(^o\) cylindrical analyzer for the plasma potential fluctuations measurements in ISTTOK is described. Preliminary estimations for the energy resolution of the ideal 90\(^o\) cylindrical analyzer indicate the value of minimal detectable change in plasma potential to approximately 5 times higher than the expected plasma potential fluctuation level. In addition, the ideal 90\(^o\) cylindrical analyzer is characterized by finite first order angular aberration. The SIMION code has been used to investigate the properties of real 90\(^o\) cylindrical analyzer with inclusion of fringing fields. As a result of numerical simulations, the necessary energy resolution has been obtained in novel approach of operation in combining electric field deflection and deceleration with biased exit electrode. Almost the first order focusing in the range of ±4\(^o\) has been obtained exploiting the effect of fringing field together with optimization of the initial conditions at the beam entrance (angle and shift relative to standard trajectory). Also, multiple cell detection at the analyzer exit is discussed as an alternative to the standard split-plate operation.
Dependence of LIBS spectra on the surface composition and morphology of W/Al coatings

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Laser induced breakdown spectroscopy (LIBS) is a promising tool for remote monitoring of erosion/deposition processes at the first wall of ITER. Proper application of LIBS requires knowing the ablation rates of co-deposited layers on plasma-facing components accurately to obtain elemental depth profiles of different elements on the layers from the recorded LIBS spectra. This goal is, however, complicated by the fact that the ablation rate depends on the composition of the layer as well as on its density, morphology, and the phase structure. To clarify the role of each of these parameters, samples with ITER-relevant 2 mm W or W/Al coatings of on Mo and without deuterium content, were prepared by three different procedures and tested by LIBS; here Al was used as a proxy for Be. In the LIBS measurements, the fluence of $\sim 7 \text{ J cm}^{-2}$ with a Nd:YAG laser ($\lambda = 532 \text{ nm}$). Time-gated spectra were recorded in the 387-410 nm wavelength interval. The LIBS results were compared with those obtained by scanning electron microscopy (SEM), secondary ion mass spectrometry and X-ray diffraction techniques. The total intensity of the LIBS spectrum recorded for the coating differed from that recorded for the substrate. In addition, SEM pictures showed that the difference became even larger as the porosity of the sample increased. This finding indicates a more efficient absorption of the laser radiation. Samples of a fixed composition but prepared by different procedures had remarkably different porosity-caused ablation rates: for W/10\% Al coating the ablation rate determined from the LIBS depth profiles changed from 50 to 500 nm per laser shot. Effect of other factors like the phase structure of the samples, was negligible.
Improving accuracy of Penning gauge spectroscopy for the determination of hydrogen isotope H/D ratios

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Deuterium-tritium gas mixture will be used as fuel in future fusion devises like ITER. Thus it is important to monitor hydrogen isotope ratios not only in fusion plasma and in the subdivektor/exhaust gases but also retained in the plasma facing components (PFC). Residual gas analysis is traditionally used to quantify the isotope species of the PFCs in the laboratory by means of thermal desorption spectroscopy (TDS). The drawback of this method is that the mass peaks of the isotopes cracking patterns and helium superimpose and complicate data analyses as well as accurate quantification. Atomic spectral lines emitted from a Penning discharge are used to quantify partial pressures and isotopes ratios in gases. To identify the potential of this method for TDS studies, the hydrogen emission spectrum lines (Hα and Dα) were examined by Alcatel type Penning gauge. The hydrogen/deuterium pressures were measured by both a membrane vacuum gauge and the Penning gauge. Different gas mixtures were produced by varying of hydrogen/deuterium flows. The Hα and Dα Balmer series lines intensities were recorded with help of a high etendue spectrometer coupled to the Penning gauge using relay optics together with fiber bundle and equipped with Peltier cooled CCD camera. Subsequent measurements using hydrogen and deuterium gases revealed for identical pressures in the range of 10^{-7}-10^{-3} mbar that the Hα line intensities are systematically higher by a factor of 1.25 with respect to the Dα line intensities. This observation can be explained by the dissociative excitation of the hydrogen molecules which has been found previously in electron beam excitation experiments [1]. Results of Hα/Dα line intensity measurements for different gas mixtures and pressures will be presented and a more accurate approach of isotope ratios determination will be discussed. [1] C. Karolis, E. Harting, J. Phys. B: Atom. Molec. Phys. 11 (2) (1978) 357
Multichannel measurement system for extended SXR plasma diagnostics based on novel radiation-hard electronics

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This work refers to the currently being developed extended soft X-Ray plasma diagnostics system with the novel, radiation-hard generation of electronics and implemented algorithms. The system is based on the Gas Electron Multiplier detector. For the multichannel, modular systems working with very intense plasmas (e.g. laser generated plasma, plasma fluxes), the phenomenon of the coinciding signals from the GEM detector can occur. It is difficult to compute energy values from events of this type with basic set of algorithms. Therefore it is necessary to perform extended verification of the incoming signals and algorithms output. In order to efficiently develop new, advanced data processing path, large amount of diagnostic data (e.g. raw signal registration, signal saturation counters) must be gathered for further investigation. This is especially difficult when working with large number of analog input channels and high event rate from the GEM detector. One module of the system is capable of processing up to 64 analog channels in real-time, while the whole unit can work with maximum number of 2048 channel. The system is designed to process plasma bursts at approximately 1 Mevents/s rate. The required bandwidth for data processing and transmission is very high. The system uses PCI-Express Generation 2 links with DDR3 memory for data storage and buffering. Artix7 FPGAs mounted on the electronic modules performs signal acquisition and data preprocessing. The extended diagnostics modules implemented in the FPGAs allows to compute photons energy together with position and corresponding analog signal. Data can be sent in real time to the PC unit for post processing including cluster construction and spectrograms computation in offline mode. It is novel approach to the plasma diagnostics and signal processing. The functionality is especially useful for development of new coinciding signal separation algorithms, which will be used in SXR plasma diagnostics systems.
Advanced probe for transport measurements in medium-size tokamaks

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The characterization of outward filamentary plasma transport in Medium-Size Tokamaks (MST) is an important objective of current fusion plasma research. We aim at improving the diagnostic of transport events in the Scrape-Off Layer (SOL) and further inside by means of various types of newly developed electrical probes combined with the associated probe measurement procedures. Presently, a New Probe Head (NPH) for measurements in the SOL of MSTs is under development. This probe head will be mounted on the reciprocating probe manipulators of MSTs. One important goal of the NPH is to introduce electron emissive probe (EEP) measurements into MSTs. If this task is successful, it is planned to construct further probe heads for tokamaks and stellarators, equipped with several EEPs along with additional diagnostics. The NPH, being the main topic of this presentation, will be equipped for measuring several plasma parameters simultaneously, like electron and ion temperature (Te and Ti), cold floating potential, Vfl, and plasma potential, Vpl, as well as the ion density ni in different ways so that a comparison of various measurement techniques will be possible. Local magnetic fluctuations will also be measured by means of miniaturized pick-up coils mounted inside the probe head. On a μs timescale, it will also be possible to compare the fluctuations of the floating potentials of two cold Langmuir probes with the EEP’s floating potential, which is expected to be close to or at the plasma potential. The results of the probe measurements are supposed to complement the other plasma diagnostic methods in MSTs delivering more information on filamentary plasma transport. The poster describes first results of an EEP prototype in a test plasma chamber and discusses possibilities of the NPH and why these measurements are important and interesting.
Coinciding signals estimation for high flux radiation in GEM detector for fusion plasma imaging

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The measurement system based on GEM - Gas Electron Multiplier detector is developed for X-ray diagnostics of magnetic confinement tokamak plasmas. The multi-channel setup is designed for estimation of the energy and the position distribution of an X-ray source. The main measuring issue is the charge cluster identification by its value and position estimation. The fast and accurate mode of the serial data acquisition is applied for the dynamic plasma diagnostics. The samples of the ADC – Analog-to-Digital Converter which are triggered by the detector current are acquired independently for the measurement channels. The FPGA – Field-Programmable Gate Array based system performs the basic functions of data processing: data receiving, signals selection, charge estimation and memory operation. High flux radiation cause the problem of coinciding signals for cluster charge identification. The amplifier with shaper determines time characteristics and limits the pulses frequency. The essential assumption is that ADC overlapping signals can be reconstructed if primary GEM pulses do not coincide. The ending tail of the signal can be restored for the given electronics characteristics. The proposed algorithm can be apply iteratively for series of superimposed pulses. Separation of coincided signals was introduced and verified for simulation experiments. On line separation of overlapped signals was implemented applying the FPGA technology with relatively simple firmware procedure. Representative results for reconstruction of coinciding signals are demonstrated. Radiation source properties are presented by the histograms for selected range of position, time intervals and cluster charge values corresponding to the energy spectra.
Development of GEM detector for tokamak SXR tomography system: preliminary laboratory tests

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Necessity to develop new diagnostics for poloidal tomography focused on the metal impurities radiation monitoring, especially tungsten emission, has become recently inevitable. Tungsten is now being used for the plasma facing material on many machines, including on the WEST project, where an actively cooled tungsten divertor is being implemented. This forced a creation of the ITER-oriented research programs aiming to effectively monitor the impurity level of tungsten in plasma. The situation is even more complicated as, due to interaction between particle transport and MHD activity, such impurities might accumulate which could lead to disruption, especially, in case of long pulse tokamaks. Therefore, an appropriate diagnostic tool has to be developed which will not just monitor the level of impurity but will also reconstruct its distribution. Combining the spectral information on plasma radiation with good spatial resolution of its detection should allow recovering fundamental information in order to estimate the level of the plasma contamination and consider its effects on plasma scenarios. Detection system based on Gas Electron Multiplier technology has been recently proposed to be used as SXR tomographic system for ITER-oriented tokamaks and is under development by our group. This work presents the current status of design of the detecting system for poloidal tomography to be installed at WEST project tokamak for the verification of the detecting concept. The detecting system consists of two detectors which are expected to be installed in a poloidal section of the WEST project tokamak – one of planar and other of cylindrical geometry. In order to study the characteristics of the detectors and verify the proposed design first laboratory tests of the constructed detectors were performed. The results of the laboratory measurements with the 55Fe source and X-ray tube will be shown demonstrating various detector characteristics such as energy, spatial and time resolution.
Validation of $^{23}$Na(n,2n) cross section as neutron flux monitor in D-T fusion reactors

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The intended fusion reaction for ITER project is $D + T \rightarrow ^4\text{He} (3.5\text{ MeV}) + ^0\text{n} (14.1\text{ MeV})$, which produces high energy neutrons. Portion of these neutrons is effectively captured in breeder blanket, however, many neutrons leak and can cause radiation damage. Monitoring of the neutron damage in ITER internals is necessary due to the aging management. $^{23}$Na(n,2n) reaction has been selected among many possible, thank to its sufficiently high threshold (12.96 MeV, in ENDF/B-VII.1), making it sensitive to fusion neutrons and insensitive for scattered neutrons with lower energy. Reference neutron spectrum is crucial point for determination of nuclear parameters in investigated elements. In case of Research Centre Rez, the reference spectrum was generated in special core assembled in LR-0 fission reactor due to very good description and many validations performed. Unlike D-T reaction, fission produces neutron spectrum with mean energy of 2.5 MeV, however still about 0.14 % of neutrons have got energy over 10 MeV. Despite the very high reaction threshold, the integral cross section has been determined to be $0.91 \pm 0.02 \mu\text{b}$, or $26.6 \text{ mb}$ in fission spectrum above 13 MeV. Comparison of the experiment with computer simulation using available nuclear data libraries shows distinctive discrepancies reaching up to 40 % in case of ENDF/B-VII.1. The data from International Reactor Dosimetry and Fusion File show difference -7.7 %. The best agreement yiealds ROSFOND-2010 with -1.9 %. The first step of validation works on $^{23}$Na has been performed and it can be concluded, the $^{23}$Na(n,2n)$^{22}$Na is suitable flux monitor for fusion devices. The research works in well described spectrum are necessary step before validation of $^{23}$Na in Gradel neutron generator, acting as volumetric neutron source, which will be used in next phases of fusion research in RC Rez.
Electromagnetic properties of some comercial REBaCuO superconducting tapes considered for magnets of fusion reactors

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Superconducting RE-BaCuO tapes of different suppliers were tested by magnetic induction (vibrating sample magnetometer, VSM) and by current transport techniques. The tests aimed at finding the best candidates for the tape utilization in a new generation of superconducting magnets for fusion reactors. The electromagnetic characteristics of the tapes as a function of temperature, magnetic field, and its angle to the tape plane were investigated. One important question was the impact of neutron irradiation on the tape properties. Therefore, the tests were made before and after a series of neutron irradiations done at LVR-15 fission reactor. The induction and transport tests provide complementary results giving us a deeper insight into the tape behavior. The induction tests enable study of vortex pinning up to very low temperatures and very high magnetic fields, where transport measurements become difficult. The transport tests, though in our case limited to the temperatures around 77 K and magnetic fields to 1 T, give information on the current flow through a long length of the tape and enable reliable tests of the angular dependence of the transport current. In contrast to the induction method, the transport experiment is insensitive to the type of the substrate and tape cover.
Influence of twisting strain on the Bi-2212 tapes and round wires under background field

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As the only high-temperature superconductors (HTS) that can be made into round wires without anisotropy, Bi-2212 has significant potential applications as CICC (cable in conduit conductor) for large-scaled superconducting magnets in fusion reactors. However, Bi-2212 is brittle and sensitive to strain which leads to a low mechanical performance. The effort on studying the impact of strain on the transport properties is mainly focus on the axial strain. During the CICC fabrication, the conductor will be inevitably subjected to twisting strain, while the twisting strain of Bi-2212 is rarely reported. The twisting strain has a large impact on CICC design and manufacturing. In this paper, twisting degradation at 77 K under background field for Bi-2212 tapes and round wires is investigated. The results show that the critical current of Bi-2212 tapes depends strongly on the magnetic field angle, while round wires have no anisotropy. The twisting strain has great impact on the critical current of Bi-2212 tapes, while less influence on the Bi-2212 tapes. Bi-2212 round wires exhibit strong advantages over tapes.
Soft real-time analysis of ITER magnetics streaming data using SPECTRE

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The SAGE ² European Horizon 2020 project (grant agreement 671500), led by Seagate with 10 partners, is investigating the needs of future exascale storage systems for data intensive applications. CCFE is one of the partners and SPECTRE (SPECtral Research Engine) is one of the tools being developed to take advantage of the improved data I/O and throughput capability of the SAGE platform. SPECTRE is designed to emulate and process the levels of data acquisition required to ingest magnetics data once ITER is running. ITER will have about 200 magnetics sources each generating data at a rate ~40 MB/s giving an input of ~8 GB/s for the whole pulse. A typical pulse will last hundreds of seconds and create over 1 TB of magnetics data. Typically magnetics data are stored and processed off-line between pulses to generate diagnostic data vital for the preparation of the next experiment. SPECTRE, using streaming technologies such as Apache Spark or Apache Storm in concert with the SAGE platform, will greatly reduce the time needed to process this diagnostic data. In addition to the significant inter-pulse analysis speed up using SAGE & ‘Big Data’ technologies, it is possible to provide soft real-time (i.e. as close to real-time as feasible) diagnostics from live streaming data – an essential aid for experiment ROs in running and guiding the pulse. Synthetic data generated at levels matching those of ITER have been used to stress test SPECTRE and demonstrate the potential benefits of the SAGE platform and related technologies. Additionally, real data generated using prototype ITER data acquisition hardware being developed by F4E, CCFE and others ³, will be analysed.

² [Link to SAGE project](http://www.sagestorage.eu/)
Status of the development of diagnostic pressure gauges for the operation in ITER

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The ITER Diagnostic Pressure Gauges (DPG) shall provide the measurement of the neutral gas pressure, which is an important parameter for basic control of the operation of ITER machine as well as for input to physics models of the plasma boundary. The reference sensor is a hot cathode ionization gauge, which is able to operate in an environment with strong magnetic fields (up to 8 Tesla), strong and noisy background signals and fast pressure changes. In total 52 DPG sensor heads will be installed in 4 lower ports, 4 divertor cassettes and 2 equatorial ports. The DPG system is currently being developed by IPP in collaboration with F4E and Sgenia within a FPA. As part of the system level design (SLD) activities several different architectures of the DPG system, including gauge head and supporting electronic equipment, have been evaluated in detail with the aim to fulfill technical requirements imposed by the integration in ITER. Design solutions developed during the SLD phase have been supported by dedicated modelling activities, which allowed assessing the impact of loads and system performance. As a result of evaluating the proposed architecture options it was decided to choose for the baseline design a system based on the technology of the ASDEX pressure gauge with the addition of implementing a thermocouple for precise calibration, implementing a baffle for thermalizing fast neutrals and optimizing the electronic equipment according to space availability inside the ITER complex, while reducing cable lengths as much as possible. For this option risk and RAMI analyses were carried out leading to the identification of 130 risks and a mitigation plan for the critical ones. Also, for the minimal operation scenario an availability of the diagnostic of 99\% for the life-time of ITER was found.
Experimental and numerical studies of the shutter dynamics for the ITER core CXRS diagnostic

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The shutter mechanical concept for the ITER core plasma CXRS Fast Shutter is based on elastic bending of a deformable arm structure (length ≈ 1.8 m) which blocks or opens the path of plasma emitted light aiming at the diagnostics first mirror. Bending of the shutter arms is induced by an actuator and will be restrained using the limiting bumpers, where, although the arms are preloaded against the bumpers, the mechanical shock excites arm rebounds followed by free oscillations. A simplified parametric shutter mock-up is used as an experimental test device for evaluation of the dynamic behaviour. Since the dynamic response of the shutter is strongly dependent on the bumper system and the pressure evolution of the pneumatic actuator, in-situ measurements are essential to study the interaction of shutter, actuator and bumper as one system. The purpose of the parametric mock-up is to investigate the dynamic behaviour by means of a simplified model where the effect of parametric changes on the key natural frequencies important for functional movement can be calculated analytically. Furthermore, the parameters of mechanical dynamics (preload, stiffness and mass distribution) are used to adjust the dynamic behaviour for optimisation during the experimental tests. Goal of the shutter dynamics adjustment is to minimise arm rebound amplitudes and duration, which can be reached through low impact kinetic energy and also by appropriate bumper material selection. The results of experimental tests and dynamic numerical simulations are compared and used for optimisation of the shutter dynamics. Beside the experimental and numerical studies, the outline of the mock-up test programme, including an experimental setup with laser displacement sensors, fast camera and load cells, is presented in this paper.
The PBS55 Upper-port Wide Area Viewing System (UWAVS) provides real-time, simultaneous visible and IR images of the ITER diverter region via optical systems located in the upper port plugs of the ITER vacuum vessel. Wall temperature and radiance measurements are performed based on the IR-images. Due to mirror contamination with reactor material deposits the optical performance will deteriorate during operation. As a result the imaging and temperature measurements will be compromised. To recover performance, a mirror cleaning system will be implemented. Cleaning optical surfaces with ion fluxes produced in a gas discharge plasma is considered as a preferred method for the ITER UWAVS first and second mirrors cleaning. Selective ion energy sputtering process with the appropriate ion flux and energy can remove contaminants while preserving the mirror optical surface. For specific conditions, threshold energies and sputtering rates are yet to be determined. To demonstrate a cleaning effect, a representative experimental prototype is developed. It reproduces the part of the Front End Optical Tube of the UWAVS diagnostic system in a 1:1 scale. The experimental prototype surfaces simulate the first and second mirrors with correct angle and separation. The surfaces hold coupons of desired material. The prototype is installed inside a larger vacuum chamber kept at low pressure mimicking the tokamak chamber. Both vacuum volumes are connected through adjustable leak valves to set the pressure for RF plasma ignition in the selected gas. Ion energies, fluxes and sputtering rates are investigated in the capacitively coupled 13.56 MHz CW and PP RF discharges and at higher RF frequencies up to 80 MHz. Experiments run in Helium, Argon and other gases at pressures between 1-50 Pa. Ion fluxes and energies are measured with the compact ion energy spectrometer placed at the high voltage RF electrode.
New piezoelectric valve for disruption mitigation studies at ASDEX Upgrade

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A disruption is a major plasma instability that follows a sudden loss of plasma energy. During such an event, large electromagnetic forces and high heat loads occur, as well as electrons at relativistic speed. These effects can cause damage to the plasma facing components and thus have to be mitigated. For this purpose high speed gas valves are used to inject a strong pulse of noble gas onto the plasma, shortly before it disrupts. Most of the plasma’s thermal energy is radiated, preventing highly localized heat loads. Electromagnetic forces are decreased due to a fast decaying plasma current. Relativistic electrons, if generated, can be dispersed before they lose confinement by injecting high-Z gas into the beam. A new valve for in-vessel high field side injection has been developed for ASDEX Upgrade. In the idle state, the gas reservoir (42 cm³) of the valve is sealed pressure-tight by the valve plate which is pressed into the Viton sealing by a steel bellow. The reservoir can now be filled with mitigation gas up to 50 bar. If the valve is triggered, a voltage of 200 V is applied to two piezoelectric stack actuators which expand immediately by a length of 0.07 mm. This stroke is amplified through a monolithic titanium frame by a factor of 30, while thereby reducing the force of the actuators and maintaining their linear behavior. Additionally, the frame serves as preload spring for the actuators. The valve stem and the valve plate are pulled back within 2 ms, opening the valve orifice, which has a diameter of 14 mm. This allows a maximal flow rate of $8 \times 10^4 \text{ Pam}^3/s$ after 1.8 ms and an average flow rate of $2 \times 10^4 \text{ Pam}^3/s$ over the total evacuation time of 10 ms. A detailed characterization will be presented at the conference.
Pressure gauge filament for neutral gas density measurement using alternating current as source power

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In plasma fusion research the neutral gas density is usually measured using hot cathode ionisation gauges which are modified for the application in high magnetic fields and for a measurement range between $10^{-3}$-3 Pa and 20 Pa. For obtaining sufficient electron emission, high temperatures in the order of 1800 K are required and thus high usually direct currents for heating. To compensate for the induced Lorentz-forces, the filament must be relatively thick to provide sufficient mechanical stability which implicates increases of heating currents. The heating current could be reduced by using a thinner filament in combination with alternating current with suitably chosen frequency to reduce mechanical stresses. To estimate the suitability of such a solution a feasibility study by means of numerical methods has been carried out. The main subject of the investigation was the hot-filament for which alternating current has been used as power source. The geometry of the pressure gauge filament used in the analysis is based on the ASDEX pressure gauge which operates with direct current. This paper provides first of all the main guidelines and features important in developing a pressure gauge filament heated by alternating current from the mechanical point of view. Exemplary, two different filament designs have been evaluated. Concerted multiphysics numerical analyses needed for this development are presented. The analyses, beginning with thermomechanical followed by electrodynamic and finally structural dynamic analysis are described. For these analyses the commercial numerical analysis package ANSYS including the MAXWEL extension have been used. Finally, the important issues for developing such a filament conceptual design like fatigue, creep, evaporation and aging of the filament are discussed as well.
Experiment of Joints Resistance and Critical Current of Bi-2212 Conductor

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Bi\textsubscript{2}Sr\textsubscript{2}CaCu\textsubscript{2}O\textsubscript{x} is a potential material for the superconducting magnets of the next generation of Fusion reactor. A R&D activity based on Bi2212 wire is running at ASIPP for the feasibility demonstration of CICC. One sub-size conductor cabled with 42 wires was designed and manufactured. A test method was designed and performed to measure the joints resistance and critical current of the Bi2212 CICC in liquid helium. A 20kA superconducting transformer, which consisted of two concentric layer-wound superconducting solenoids, was used to provide the current of the conductor sample. Both of the primary and secondary coils were immersed in liquid helium during the experiments. The highest current of the secondary loop was up to around 14kA. The critical current of the conductor was 13.2kA with criterion of 1\textmu V/cm and the joints resistances were around 20n\Omega.
Standardization of the hard- and software used to operate manipulators at ASDEX Upgrade

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Manipulators are an important tool to position diagnostics or samples near to the plasma without breaking the vacuum of fusion devices. They can be used for different purposes like measuring plasma parameters with electrical or magnetic probes near to the core plasma or to investigate plasma-wall interaction by exposing dedicated samples. ASDEX Upgrade is operating a set of manipulators, the midplane manipulator, the divertor manipulator, the reciprocating x-point probe and two fast ion loss detectors. These manipulators were developed and installed over more than 20 years of operation of ASDEX Upgrade. The hardware and the control systems are based on different platforms. A refurbishment of manipulator systems required for various reasons was used to implement a standardized hard- and software. The current divertor manipulator and the reciprocating x-point probe are completely new constructions and only located at the same place as their predecessors. The old control systems could not be used for the new designs and therefore new control systems had to be built up. The midplane manipulator is more or less unchanged regarding its mechanical setup, but it got a new control system with matching new motors. The new control systems are designed as similar as possible, regarding both maintenance and operation. The manipulators are now driven by servo motors and motor controllers of the same model family, combined with S7 Simatic PLC controls for interface and also to control the peripheral systems like the generation of the vacuum and to connect the manipulators to the ASDEX Upgrade control system. This paper gives an overview about the status and recent updates of the different manipulators and their control systems, their capabilities and their integration into the ASDEX Upgrade framework.
P4.084

The irradiation damage assessment on TF coil in CFETR with HCSB blanket

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Chinese Fusion Engineering Testing Reactor (CFETR) is a test tokamak reactor to bridge the gap between ITER and future fusion power plant and to demonstrate generation of fusion power in China. In order to select the most suitable blanket proposal for CFETR, the three blanket concepts (i.e. the helium cooled solid breeder blanket, the liquid LiPb blanket, and the water cooled ceramic breeder blanket) are under development and evaluation simultaneously. A Helium Cooled Solid Breeder (HCSB) blanket was proposed by School of Nuclear Science and Technology, University of Science and Technology of China, and its conceptual design has been carried out. In this paper, the specific issues and analysis on damage to TF coils of CFETR with HCSB blanket were carried out based on the three-dimensional model of the CFETR with the widely used code MCNP and the IAEA latest released FENDL/2.1 data library. Damage to some specific regions of the TF coils near large openings and at the inboard mid-plane are calculated and analyzed. Parameters such as the distributions of nuclear heat density, fast neutron flux, dose rate to the epoxy insulator, and peak displacement dose to Cu conductor for the TF coil near these regions were calculated and analyzed. The shield thicknesses at these regions are optimized. Keywords: Irradiation damage, neutronics, TF coil, CFETR, Helium cooled solid breeder
Progress in the new EAST magnet feeders

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The original EAST magnet feeders have been operated for over 7 years since 2006. With the improvement of experimental parameters, a new magnet feeder system has been designed for the upgrade project of the EAST. It consists of 13 pairs of superconducting bus-lines with total length over 900 m and 13 pairs high temperature superconducting current leads. Each original bus-line connecting new cable in conduit conductors by twin-box lap joints, is extended to 22 m. A pair of 16.5 kA current leads are designed and fabricated. The insulation resistance between the TF bus-lines and the ground after the installation is 450 MΩ, and the resistances of the PF bus-lines are over 1500 MΩ. The average DC resistance of the 26 twin-box lap joints is below 5 nΩ after the feeders being cooled to the operation temperatures. The total heat load into the 5 K zone in bus-line groups was calculated to be less than 0.3 W/m. Three physics experiments from 2014 have demonstrated successfully the new magnet feeder system is stable in the operation.
X-mode raw data analysis of the new AUG ICRF antenna edge density profile reflectometer

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The new multichannel X-mode reflectometer installed on ASDEX Upgrade measures the plasma density profile evolution at different positions in front of the ICRF antenna. The reflectometer operates in the extended U-band (40–68 GHz) microwave region, measuring density profiles up to $10^{19}$ m$^{-3}$ with magnetic fields between 1.5 T and 2.7 T. In this heterodyne reflectometer architecture, the signal reflected by the plasma is downshifted to a lower intermediate frequency, amplified and filtered with a 100 MHz bandpass filter. Quadrature detectors demodulate the in-phase and quadrature (IQ) signals, which are acquired at 200 MS/s. In this work we analyse the acquired IQ signals from the different reflectometer antennas, and describe the waveguide dispersion calibration and filtering of the raw signal. The effect of spurious reflections, such as the multiple reflections from the ICRF antenna metal straps, are analyzed and taken into account on the data processing software. The amplitude and phase characteristics of the signal with plasma reflection are used to determine the first fringe of the upper X-mode cutoff and calculate the group delay used for the density profile inversion. In some high plasma density and high magnetic field scenarios, both the lower and upper X-mode cutoff frequencies are detected in the single U-band. An algorithm to distinguish both cutoff regions is presented, enabling the use of the lower cutoff group delay for the future implementation of the core density profile inversion.
P4.088

Automatic pattern recognition on electrical signals applied
to Neutron-Gamma discrimination

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The electrical pattern recognition can be useful in several applications, generally it is used to detect particular events or anomalies in the signal under analysis or to identify precursors, especially in electrophysiology. Each application requires customized algorithms and appropriate signal processing capabilities. In this paper we present an application of pattern recognition to real-time discrimination of neutrons and gamma rays detected by liquid scintillators; the discrimination is possible because the two particles incident the detector produce pulses having different shape. A general-purpose algorithm is proposed that can be efficiently implemented in a programmable logic gate array; this allows the development of efficient and low-cost systems for the electrical pattern recognition which, with minor changes, can be applied to different diagnostic fields. The discrimination of particles is performed starting from a reference patterns set. This reference set can be simple and with a limited number of patterns; however the hardware implementation may result complex, due to the high bandwidth of the signals under analysis. The proposed pattern recognition algorithm is based on the cross-correlation operator and on the definition of a norm related to the difference between the reference pattern and the shape of the actual signal. The automatic pattern recognition algorithm, the digital hardware implementation, its software, as well as the simulations done in case of general purpose patterns and are described in the paper. Moreover, in order to verify the performances in the case of scintillator signals, the algorithm has been applied on data acquired by a scintillator system irradiated by a neutron-γ source at the Frascati Tokamak Upgrade laboratories. The results confirm the suitability of the method and its future usability.
An assessment of axial pre-compression for the KSTAR central solenoid


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The central solenoid (CS) of the KSTAR consists of four pairs of superconducting coils compressed axially by preloading structures. The axial pre-compression was designed to 15 MN at 5 K, which could suppress the maximum repulsive force of the coils based on reference operation scenarios. Tolerances in-between insulations, buffers, wedges, blocks and shells have been precisely controlled during the assembly of eight winding packs. Only shells of preloading structures were heated to make 6 mm gap between the preloading structure and winding pack assembly. The gap was, then, filled up by wedge movement so that axial pre-compression be applied after thermal contraction of shells. However, measured pre-compression is about 60% of the design value. A detailed investigation has been carried out to find reasons for the preloading reduction by comparing strain measurement and structural analysis results. Our study on the axial pre-compression for the CS is expected to give a guide line for future KSTAR operations.
P4.091

Optimizing BUSSARD, the new 16-phase inverter system of ASDEX Upgrade

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BUSSARD is a new inverter system at the nuclear fusion experiment ASDEX Upgrade for mitigation of ELMs and execution of other, physics related experiments. The concept and first results were presented in detail [1]. Four-phase operation was routinely done during shot campaign 2015/16 and many experience in operation was gained. Now, the completion of BUSSARD is almost finished and many improvements were adapted. 16-phase operation with up to 16x 1.3 kA coil current of arbitrary waveform controlled by 16 independent real-time controllers at 500 Hz bandwidth, 5 kHz switching frequency and about 1 MW/10 Mvar total real/reactive power will be commissioned within the current campaign. In this publication, three main topics are discussed: (i) Simulations, (ii) re-design of interface cards and (iii) the new in-house developed GUI (graphical user interface) for efficient handling of BUSSARD. Different kinds of SPICE-based models were developed and verified with experimental data. A “global model” describes the whole electric system with focus on power distribution and power circulation between sources and loads. Here, the 16 inverters can be fed by realistic reference curves, but they are approximated by simplified electrical models. This reduces calculation time and the model is helpful for quick trouble-shooting, optimization of main thyristor rectifier controllers and identification of valid working regimes. Another model describes the behaviour of quick switching power electronics and its electrical environment in detail. Such model requires more computing resources but it was important during design phase of BUSSARD. A third model was developed for improvements of PWM scheme and fast real-time controlling algorithms. The re-design of interface cards comprehends additional disturbance suppression, optimization of grounding schemes and potential separation, improvements in fail-safe handling of critical safety-relevant signals and optimizations of PLD-related firmware. [1] M. Teschke, et al., Fusion Eng. Des. (2015) 171–176
P4.093

The DC-link of the inverter system BUSSARD for ASDEX Upgrade in vessel saddle coils

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Recently an inverter system (called BUSSARD) was assembled to individually feed the 16 in-vessel saddle coils of the fusion experiment ASDEX Upgrade (AUG). The new inverter system consists of 16 inverters, each with an output current of up to 1.3 kA and a bandwidth of up to 500 Hz in arbitrary waveforms. Currently, the system is in operation with 4 inverters feeding four in serial connected coils, each. The full system will be commissioned in 3/2016. The common DC-link is fed by an existing thyristor current converter, called “Group 0”. The standard operation of this converter is current control mode which remains needed in the future for other applications. But for feeding of BUSSARD’s DC-link, voltage control mode is required. Therefore the control concept of the thyristor rectifier had to be modified. During the development process several control concepts were designed, simulated and tested. For this application the feed forward control was chosen as the best solution with respect to stability. The Group 0 consists of two independent 2-quadrant thyristor rectifiers (module 0.1 and 0.2). Each module can provide a current of up to 4kA at a voltage of 340V or 600V (star- or delta-circuit configuration) for up to 10 seconds. With the existing output inductances the DC-link voltage oscillates and has strong voltage drops during quick load changes. To avoid this, new inductances, which are well-adapted to the capacitance of the DC-link, were integrated into the output of the modules. For a further optimization of the DC-link voltage level and ripple, some investigations on the real power distribution were done. All steps required to optimize the converter for BUSSARD’s DC-link supply including results of the first operation campaign are presented in the paper.
Study on 300MVA pulse generator starting system

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Study on 300MVA pulse generator starting system. For supplying power for HL-2M Tokamak, a new 300MVA pulse generator has been developed. The new generator with 400 tons of rotor to stored energy will be driven by an 8500kW asynchronous motor. The purpose of the research is how to reduce the startup current and start the motor steadily. One way is to use liquid resistors in series with the motor rotor. Simulation and calculation are made for motor rotor series of liquid resistors. The maximum series resistors, the starting current and starting time are obtained. And the cooling capacity to cooling liquid resistors and the temperature rise are achieved. Series H-bridge multilevel high voltage varying frequency starting of winding asynchronous motor is another kind of high efficiency way to startup. In this paper, the cascade multilevel converter working principle is analyzed. Multiple transformation rectifier circuit is theoretically proved to eliminate lower than 6m±1 harmonic current. A simulation model of cascade multilevel inverter PWM control is built by the MATLAB/SMULINK, and pulse phase shift PWM control scheme is analyzed. The results show that the pulse phase shift PWM is a kind of multilevel PWM control scheme of the practical project. Based on the abovementioned study, a starting equipment of rotor series of liquid resistors and a 6000V / 12000kVA high voltage variable frequency system are designed for 8500kW motor. The main circuit is introduced. The test voltage and current waveform of the device are given, and the theoretical analysis and experimental results are basically consistent. At present, two sets of equipment are installed and debugging to be finished.
P4.095

Development of 300 MVA Motor Generator in HL-2M

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A new motor generator (MG) system is building mainly for the poloidal field power supply system of the HL-2M Tokamak. This MG system will be capable of providing a peak capacity of 300 MVA and delivering up to 1350 MJ per pulse at 15 min intervals. The system consists of a 300 MVA MG and its auxiliary systems. The MG adopts the semi umbrella vertical shaft type and consists of an 8500kW induction motor and a 6-phase synchronous generator. The basic specifications for the generator are peak output of 300 MVA at 0.65 power factor, rated current of 28,868 A, rated frequency of 99.6 \textasciitilde 67 Hz, and rated work speed of 498 \textasciitilde 335 rpm. The optimized electromagnetic parameters of the MG is given out by studying the MG and the desired requirements of its load. The performance analysis of the MG by FEA shows that the fatigue life, the stress and the temperature rise can meet the special requirements of the MG operation. All of the MG components had been machined and the installation was started in the second half of 2014. The rated speed of the MG is high, the speed decreases rapidly per pulse and the rated current of the generator with double Y stator windings is very large, so high quality control is necessary, including the rotor eccentricity of \leq 0.15mm and the whole shaft verticality less than 0.01mm/m are strictly kept. Many checking tests were carried out during the installation. At present, the static debugging of the installed MG is performing for its first start-up test.
Development of a quasi-steady state high voltage power supply based on Marx generator

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A high voltage power supply (HVPS) used for the ECRH system on the SUNIST tokamak is introduced. It is able to output a 50 ms pulse of -40 kV / 15 A in every 5 minutes. The voltage drop for the whole flat top is less than 2%. In each arcing events, the maximum energy delivered to the load is less than 15 Joules. The HVPS is based on Marx Generator and PSM technologies using fast switch elements (IGBT), which consists of 150 identical modules. Every module contains an electrolytic capacitor (10 mF / 450 V) for energy storage, an IGBT switch, an IGBT driver circuit, and other remote control circuits. Each module are self-powered. In other words, all the low power consumption electronic circuits, including the IGBT drive circuit and other remote control circuits get power from the electrolytic capacitor. This greatly simplified the design and lowered the cost. Each module are connected in the topology of Marx Generator, which means that all the 150 electrolytic capacitors can be charged in parallel using a normal low voltage (~ 400 V) power supply. The drive signals of IGBTs are delivered by optical fibers for fast response and high voltage insulation. The HVPS utilizes the control logic of PSM technologies, which can deliver user-defined waveforms to loads and keep the voltage drop of the flat-top less than 2% for a 50 ms pulse.
High heat flux testing of newly developed tungsten components for WEST

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The WEST (W -for tungsten- Environment in Steady-state Tokamak) project is based on an upgrade of Tore Supra tokamak. ITER-like actively cooled tungsten targets (monoblocks) will be integrated in the lower divertor and a new set of actively cooled tungsten coated plasma facing components will cover a part of the vessel to provide a fully metallic environment. In preparation of the production of the actively cooled tungsten targets for the lower divertor the qualification of suppliers is done by exposing small-scale mock-ups to cyclic high heat flux tests in the electron beam facility JUDITH 1 at Forschungszentrum Jülich. Thereby, the small-scale mock-ups are based on the ITER design, consisting of 7 monoblocks mounted on a CuCrZr cooling tube and connected via a soft Cu-interlayer. The results presented herein focus on the high heat flux test results of two mock-ups produced by ASIPP in China. The testing was performed by thermal cycling at 10 MW/m² and at 20 MW/m². While for the first component failure occurred after 769 cycles at 10 MW/m² due to accidental conditions in the testing facility, the second component, being pre-exposed on two blocks to 700 cycles at 10 MW/m², survived 500 additional cycles at 10 MW/m² and up to 500 cycles at 20 MW/m² without obvious damage formation. The discussion of the results comprises the thermal performance of the mock-ups during thermal cycling in relation to the initial qualification via infrared thermography facility SATIR at CEA-Cadarache, the tungsten thickness as well as the subsequent microstructural analyses. The latter focus on the most important issues determined in earlier studies on ITER qualification mock-ups and components, i.e., tungsten recrystallization and macro-crack formation, the integrity of the CuCrZr cooling tube, and plastic deformation induced changes in the pure Cu-interlayer.
Development of reliable high heat flux removal techniques is an important issue to design plasma facing components in a fusion reactor. The ITER-like divertor cooling design based on water-subcooled flow boiling is one of the well-developed divertor cooling schemes. To withstand such a high heat flux in the vertical target of the ITER divertor, a twisted tape is inserted into a CuCrZr tube imbedded in a tungsten monoblock, which is called the swirl tube. The twisted tape generates the rotational swirl flow along with axial flow, which can effectively tear off vapor bubbles from the tube wall in water-subcooled flow boiling and thus enhance critical heat flux (CHF). However, the twisted tape increases CHF at a cost of rise in pressure drop. Therefore, designs of twisted tapes which further maximizes cooling capacity while minimizing rise in pressure drop are desired. In this study, the design and fabrication of swirl tubes with innovative twisted tapes is introduced. The conventional twisted tape has the width as same as the inner diameter of the tube, which obstructs fluid flow in the core region. On the other hand, the proposed design has the twisted rail-like tape consisted of two narrow strips spaced out at the core region, which is expected to generate swirl flow near the tube wall to remove vapor bubbles and reduce flow resistance due to the unobstructed flow area in the core region. To validate and optimize the proposed design concept, a set of CFD analysis is performed by varying main design parameters of twisted rail-type tapes. Test specimens with the complex geometry of the rail-type twisted tape are manufactured thanks to recent metal-based 3D printing techniques. Finally, CHF and pressure drop of water-subcooled flow boiling in a swirl tube with various rail-type twisted tape are experimentally investigated.
Heat flux test and cooling effect of tungsten brazed mockup with swirl tube
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It is so important that the bonding technology between tungsten and dissimilar metals for the PFC of ITER and DEMO. The development of tungsten brazing technology was first launched for the KSTAR PFC. Flat type tungsten block was brazed on CuCrZr in vacuum at a temperature of 980 °C for 30 minutes using silver free brazing alloy. A OFHC-copper was used as an interlayer between tungsten and CuCrZr because of its low yield strength and low elastic modulus. The brazing filler is a 0.05 mm thick-plate made of the Ni-Cu-Mn alloy. Tungsten brazed mock-ups with a swirl tube were tested at an electron beam facility, KoHLT-EB(Korea heat load test facility) in KAERI. The high heat flux test was performed for tungsten brazed mock-ups with a swirl and smooth tube under heat flux of about 5 MW/m\(^2\) up to 2,000 cycles and about 8 MW/m\(^2\) up to 2,000 cycles. The ultrasonic test was performed to inspect the bonding between tungsten and CuCrZr, and the microstructures of the bonded region were analysed by scanning electron microscopy after the heat flux test. The test results show there are no delaminations or failures at the bonding joints during and after all the heat flux test and the swirl tape is better for cooling the surface of tungsten mockups under high heat flux. In this study, we present the manufacturing process of tungsten brazed mock-ups with a swirl tube in detail and the results of the high heat flux test.
Thermal-lifetime Analysis of Tungsten Coated Plasma Facing Component

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Tungsten coated mock-ups for developing the Plasma facing component (PFC) were fabricated and tested in the plasma torch and high heat flux test facility with electron beam, which can be used in the repair of the damaged PFCs. For evaluating the life-time of the tungsten coated mock-up, the erosion rate was measured and thermal-lifetime analyses were performed with the fabricated mock-up. And the results were compared with the HIPped (Hot isostatic pressing) mock-ups. Thermal-hydraulic and thermo-mechanical analysis with the conventional codes such as ANSYS-CFX and ANSYS-mechanical were performed to evaluate the thermal lifetime according to the thickness of the Armour material.
High heat flux test facility KoHLT-EB and development of plasma facing components in Korea

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The Developments of plasma facing components (PFCs) are the key items for the nuclear fusion reactors. The most components for the tokamak PFCs are the blanket first wall, divertor, heating ports, and diagnostics ports. These PFCs are composed of the armour materials, the heat sink for the cooling, and the structural materials. Be, W, C-composites, and advanced materials were selected for these armour in the case of the ITER and DEMO concepts. For the development of these plasma facing components, small test mockups are fabricated and tested in a high heat flux test facility to evaluate the thermal life cycles. The high heat flux facility for these performance tests in Korea, KoHLT-EB (Korea Heat Load Test Facility-Electron Beam) has been operated by KAERI (Korea Atomic Energy Research Institute). The beam power is a 300 kW with the electron gun and mid-frequency power supply, and the allowable target area is 70 cm x 50 cm. For the high heat load test of PFCs, small mockups were fabricated by various bonding techniques. Tungsten armour mockups were 1) W and FMS HIP bonding, 2) W coating in FMS structural materials up to 3 mm, 3) W and Cu bonding, and 4) 3D metal printing cooling structure. Each fabricated mockup was installed inside KoHLT-EB, and thermo-hydraulic tests and thermal fatigue tests were performed to qualify the mockups specification and bonding techniques.
After years of exploration and development, research of magnetic confinement nuclear fusion is progressed into stage of experimental fusion reactor construction and test. As a key plasma-facing component, the anti-fatigue performance of first wall of fusion reactor receives widely concerns. Due to the fact of enduring both periodic loads of pulse operating mode and shock loads of transient events such as disruption, ELMs etc, the coupled fatigue responses of material and structure are in the state of very complex. It is significant and necessary to research the coupled mechanism of fatigue by both transient and periodic heat loads, which will be beneficial to develop the key and new technology of promoting anti-fatigue performance for the first wall of fusion reactors. With such motivations, a relative complete finite element analysis method based on a full coupled thermal/structural heat transfer equation with consideration of elastic/plastic constitutive relation as well as multiple kinds of thermal physical effects such as melting, solidification, evaporation etc. is established. With this method, the thermal/mechanical response of first wall and its fatigue performance are investigated. The results show that: (1) Heat is mainly deposited on PFM layer, leading to a mechanical irreversible damage of repeated thermal elastic and plastic expansion, contraction and yielding, the first wall with graded W-Cu PFM is potential of higher heat shock resistance performance. (2) The fatigue performance of first wall with PFM of Wu-Cu graded material is very different with different graded index parameter of material. An optimized index parameter is obtained. (3) The fatigue life time of first wall is decreasing nonlinearly with increase of heat loads magnitude and the coupled periodic normal loads and shock loads induced by transient events will greatly reduce the fatigue life time of first wall.
Design and development of high pressure high temperature water circulation system for HHFTF

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This paper deals with the design of High Pressure High Temperature Water Circulation System (HPHT-WCS) for High Heat Flux Test Facility (HHFTF) of IPR and its related thermal hydraulic experiments. HHFTF has been established at IPR, India for testing performance of plasma facing components under intense heat loads expected in plasma fusion devices. Plasma facing components of the present day Tokamaks are primarily water cooled e.g. ITER. In order to test the water cooled test mock-ups or components in HHFTF, HPHT-WCS is being established that can supply water at desired pressures, temperatures and flow rates. Apart from testing thermal load performance of the mock-ups/components, this experimental loop will also be useful for validation of the computational results obtained from CFD codes which are used to develop/investigate various new concepts of water cooled components. HPHT-WCS, a new planned experimental loop is capable of providing de-mineralized water up to 160°C temperature, 60bar pressure and 4.85 kg/s mass flow rate. The loop is designed on a skid base frame which has piping connected to various process components and the system is operated remotely through a data acquisition and control system. The main requirements and characteristics, design and development, results of performance testing of the HPHT-WCS are presented in this paper.
It is desirable to develop tungsten (W) diverter in Tokamak-type nuclear fusion reactor including the International Thermonuclear Experimental Reactor (ITER). W has the highest melting point in all metals and thus is a promising material of the diverter. Since the diverter will repetitively undergo high heat flux of 100MW/m$^2$ at least in a few tens of millisecond or less when plasma disruption occurs, it is predicted that W of the surface material would be molten and solidified during the operation. The surface shape of the W diverter after or during such repetitive heat load has the strong influence on the lifetime of it due to change of the thermal resistance. Thus, it is important to establish the method predicting the W behavior for the W diverter design. However, it is difficult to evaluate it with phase change during short time only from an experimental study. We have been developing simulation method which can predict a metal behavior with phase change for laser processing and our simulation method can be extended to the W diverter. In our simulation, equation of continuity, momentum, energy and equation of state (EOS) are solved by C-CUP method and then multi-phase phenomena can be considered. Firstly, we evaluated aluminum (Al) and nickel (Ni) behaviors irradiated by short pulsed laser using SESAME as EOS. It was confirmed that our simulation method could treat phenomena inner or outer materials such as shock wave propagation and ablation. After that, we tried to introduce GRAY-EOS in governing equations instead of SESAME in order to simulate various materials including W, because we didn’t have any database of SESAME other than Al and Ni. The simulation of W irradiated by laser is conducted using our simulation method and the result is compared with that in the experiment of Ueda et al..
Feasibility study of fabrication of large scale mock-up and strength evaluation of HIP joint

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A hot isostatic pressing (HIP) method is one of the candidate processes to fabricate the fusion blanket with built-in cooling channels. Thin plates and rectangular tubes made of reduced activation ferritic/martensitic (RAFM) steel, such as F82H, are consolidated by the HIP method. The wall quality therefore depends on the integrity of the formed HIP joint. In laboratory scale experiments, electrolytic polishing to HIP bonding surface was performed to remove the machined layer. As a result, the strength of the HIP bonding surfaces was improved. To investigate this reason, oxidations on the HIP joint were analyzed. As a result, it showed that the number of oxidations on the HIP boundary were reduced to approximately 0.5 times. So the purposes of this study are to verify feasibility of electrolytic polishing to the large-scale components and to measure the strength distribution of the HIP joints. Material was F82H steel, which chemical composition was 7.78Cr-1.99W-0.20V-0.03Ta-0.12Mn-0.11Si-0.09C (Fe balance, all wt%). The assembly was then consolidated by HIP at 150MPa for 2 hours at 1100°C. After the torsion tests, the fracture surface and cross-sectional HIP interface were observed by scanning electron microscope (SEM). The elemental analysis on the observed inclusions was conducted utilizing transition edge sensor type X-ray analysis system (µ-EDS) assembled to SEM, and field-emission electron prove micro-analyzer (EPMA). Accordingly, even for large-scale components, it was confirmed that machined layer was removed by electrolytic polishing. And without large deformation due to HIP, Mock-up has been fabricated. Then, the test pieces were machined from the mock-up, torsion tests were conducted. And this topic will be discussed further in this paper.
P4.112

Experimental simulation of super-X divertor for detached plasma by TPD-Sheet IV

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In the magnetic confinement fusion reactor for high power and long pulse operation, enormous heat flux (exceeding 10 MW/m\textsuperscript{2}) is expected to flow onto divertor plates from core plasma. In order to reduce this heat load, the divertor geometry on stationary detached plasma formation must be realized. In addition, the neutral particle flowback into the core plasma is necessary to suppress by the divertor geometry. The super-X divertor has been proposed by DEMO divertor concept, investigated using simulations. Its optimization needs detail information on confined neutral particles and their physical process. Experimental simulations of divertor geometries for the formation of detached hydrogen sheet plasma using the linear divertor plasma simulator TPD-Sheet IV\textsuperscript{1}). In order to understand the basic mechanism of detached plasma, we have carried out an experiment using Super-X target geometry. In this experiment, the ionization and recombination rate are discussed by using the Collisional-Radiative (CR) model. The electron density and temperature were measured using a Langmuir probe. Ionization and recombination ratio (RIon, RRec) are discussed from experimental data of Te and ne, using the CR model. In the result that Super-X target effectively enhances plasma detachment in the low pressure. 1) S. Tanaka, et al.: Fusion Sci. Tech. 63 420 (2013).
Progress in design and prototyping activities for the Material Plasma Exposure Experiment

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One of the critical challenges for the development of next generation fusion facilities, such as a Fusion Nuclear Science Facility (FNSF) or DEMO, is the understanding of plasma material interactions (PMI). The field of PMI occurs at the intersection of plasma physics, materials science, and engineering, and requires expertise and research and development in each of these fields. Making progress in PMI research will require integrated facilities that can provide the types of conditions that will be seen in the first wall and divertor regions of future fusion facilities. To meet this need, a linear plasma facility, the Materials Plasma Exposure Experiment, is proposed. The plasma source will be a helicon antenna, with heating provided by electron Bernstein wave and ion cyclotron heating systems. This will produce heat fluxes of up to 10 MW/m² and ion fluxes of up to $10^{24}$/m²·s over a 75 cm² area at the target. In order to provide long-pulse conditions, plasma will be confined with superconducting magnets with on-axis fields from 1-2.5 Tesla, and all plasma facing components will be actively cooled. In order to examine the plasma interactions with neutron damaged materials, MPEX will have the capability to handle low activation irradiated samples. A vacuum cask which can be disconnected from the high field environment in order to execute in-situ diagnostics is planned for the facility as well. Details of the pre-conceptual design and research and development of the various systems of MPEX will be presented. This includes the plasma and heating sources, the vacuum system, the water cooling system, the magnet and cryogenic system, the controls systems, the diagnostic systems, and the sample handling and examination stations. Progress in the design and prototype testing will also be presented.
Critical processing considerations for ITER first wall beryllium tiles

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In its current design, the ITER fusion machine will use tens of thousands of beryllium tiles as plasma-facing components in its First Wall. S-65 is one of three grades of beryllium which has been accepted by the ITER International Organization for use in the reactor. The beryllium material for ITER has to pass through many machining and manufacturing processes after being consolidated by vacuum hot-pressing in order to be made into the tiles that will be used in the First Wall. Due to beryllium metal’s inherent reactivity, pitting of the beryllium surface can occur during these processes, if it is not handled correctly. As there are many different manufacturing facilities around the world who will be producing the ITER First Wall tiles, a discussion on proper handling of beryllium through all of the tile fabrication processes will be offered. This paper will characterize the pitting that can occur and discuss its causes and the methods that can be employed to prevent it from happening.
Tritium measurement for tungsten deposition layer by imaging plate technique after tritium gas exposure

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It is important to understand tritium (T) desorption behavior from plasma-facing materials of a fusion reactor in order to discuss tritium recovery method from in-vessel components. Tungsten (W) is a candidate material for plasma-facing components. Although a sputtering rate of W by hydrogen isotopes is low, a certain amount of W deposition layer will be formed on plasma-facing wall. In this work, T desorption behavior from W deposition layer exposed to gaseous T was investigated by imaging plate (IP) technique. Samples of W deposition layer were formed on W substrates by hydrogen RF plasma sputtering. Prepared samples were separately put in a reaction tube and exposed to the T-D gaseous mixture (7.2% T/D) at 573 K or 773 K for 3 hours. Then, the closed reaction tube contained the samples was transported into a glove box filled with argon and T level of the samples were investigated by IP technique without air exposure. After that, annealing (368, 423, 573, 773 and 973 K) and following IP measurement were repeatedly performed. For comparison, the W substrates without W deposition layer were exposed to the T-D gas mixture at the same conditions. The initial intensities of photo-stimulated luminescence (PSL) were 1029 PSL/mm²/h for the W deposition layer and 81 PSL/mm²/h for the W substrate. A part of retained T in W deposition layer was released at a low temperature of 368 K. However, the majority of the retained T was removed at 773 K for the sample exposed at 573 K and removed at 973 K for the sample exposed at 773 K. These results indicate that the formation of the W deposition layer increases the T inventory in the vessel and T removal from W deposition layer by in-vessel baking at around 423 K is limited.
Study of tritium and helium generation and release from lithium-containing materials

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Tritium is a prospect fuel material for future fusion power reactors, thus tritium breeding in these reactors is one of the design challenges, which can be solved by using the lithium-containing materials for construction of the reactors’ blankets. Also of great interest is use of lithium as a plasma-facing material, for example, in the form of lithium-capillary porous systems (CPS). Such systems showed promising results during numerous experiments at plasma accelerators and under conditions of operating tokamaks. To use lithium in fusion reactors it is necessary to estimate the parameters of hydrogen isotopes recycling in fusion reactor’s chamber taking into account the processes of sorption and desorption by the chamber’s lithium surfaces. Also it is important to take into account the processes of tritium recovery in lithium under neutron irradiation. The paper provides an overview and analysis of the experimental results on study of tritium and helium generation and release from the following lithium-containing materials: lithium, lithium CPS, lead-lithium eutectic (83%Pb+17%Li), lithium ceramics (Li₂TiO₃) under reactor irradiation at the reactors IVG1.M and WWR-K (Kazakhstan). The following general patterns have been revealed during these experiments: For all lithium-containing materials a significant contribution into overall tritium yield is made by release of fast atoms formed in the surface layer due to the nuclear reaction $^{6}\text{Li}(n,_{\alpha}^{3}\text{H})$. For lithium-containing materials release of helium atoms exceeded of tritium ones, given that helium is inert, does not react with the atoms of other materials, and has a substantially activationless character of desorption with material open surface. Analysis of the temperature dependences of tritium flow from the samples showed that the most important process affecting tritium release is its interaction with lithium atoms, which results in formation (decomposition) of lithium tritide LiT.
Low temperature deuterium release from lithium films

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Lithium is considered as a promising material for plasma-facing components (PFC) in future fusion devices. A number of experiments have already demonstrated positive effects of lithization and using of Li based PFCs on plasma operation. During operation of the machine, lithium is deposited on the surrounding walls and in shadowed areas. One can expect a high concentration of hydrogen isotopes in the resulting films due to good getter properties of lithium. Therefore, co-deposition of hydrogen isotopes and lithium should be investigated. The Li-D films were co-deposited in magnetron discharge in deuterium. Deuterium accumulation in the films was measured in-situ by means of thermal desorption spectroscopy (TDS). The deuterium concentration in the films was calculated to be 10-20 at. %. The main part of deuterium released in one sharp peak with the maximum at ~710 K. It was also demonstrated that deuterium can release from lithium films even at room temperature in the case of interaction with water vapour or with air. Interaction with other atmospheric gases (oxygen, nitrogen) is much weaker and did not lead to significant deuterium release at room temperature. The dynamics of deuterium release during interaction with water vapor was also monitored by quadrupole mass-spectrometer. At the pressure of 1 Pa, significant part of deuterium released already during first several minutes. After 30 minutes of exposure almost no deuterium remained in the film.
Retention and transmission properties of deuterium in tungsten by the divertor simulator TPD-Sheet IV

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Tungsten is important candidates for plasma-facing component applications on the development of magnetic fusion reactors. Particularly, it is important to understand the behavior of hydrogen isotopes in tungsten of the diverter wall material. In this study, we have performed the irradiation experiments using deuterium and helium mixed plasma in order to investigate the deuterium retention and transmission properties in the tungsten material. In the irradiation experiment, it was carried out using the linear divertor simulator TPD-Sheet IV. Samples were positioned at the end of the plasma column. Used as samples is ITER grade tungsten in the form of square plate with the thickness 1mm, was annealed to adjust the crystal grain boundaries. The deuterium transmission property of the tungsten material was investigated by a titanium plate which is mounted behind the tungsten as deuterium storage material. The plasma was irradiated to the sample is 4 types of deuterium plasma only and the D-He mixed plasma (helium 0%, 5%, 10%, 15%). The mixing ratio of helium was measured using an omegatron mass analyzer. The retention and transmission properties were measured as function of the irradiation time (15 minutes, 1 hour, 2 hours). In addition, these properties after the irradiation is examined with a TDS. When the gas flow rate of the helium is increased, amount of deuterium in tungsten is decreased and amount of deuterium in titanium is increased. This indicates that deuterium transmission property in the tungsten are influenced by the helium gas ratio.
Virtual prototyping tools for the JET divertor

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Virtual prototyping enhances traditional engineering analysis workflow when a quick evaluation of complex load cases is required. During design, commissioning or operating phases, components can be virtually tested in realistic conditions by using previously validated numerical models and experimental databases. Three complementary applications have been developed under this approach for the JET divertor. Their aim is increasing its operational range, reliability, and understanding. At the same time, they are designed to be extensible to any plasma facing component.

ALICIA is a novel inverse code suitable for use with complex geometries and noisy input temperature signals (1D IR maps). It calculates power profiles—including ELMs—with unprecedented accuracy, defining a realistic input for the recreation of the plasma load for each pulse. This reliable data is extremely important for the development, validation and application of the other two forward codes.

VITA simulates the temperature evolution of divertor tiles using experimental conditions stored in the JET database. It can produce synthetic tile surface temperatures—accurate with respect to measurements—, which could play a critical role for operating the tokamak if divertor IR measurements were not available. It can also be used during the preparation phase to check more accurately (hence with a reduced margin) a planned pulse will remain within the temperature limits.

WHAM is the first nonlinear thermal finite element solver designed to work in a tokamak real time machine protection system. It is included as a module for the Wall Load Limitation System (WALLS), simulating the transient 2D thermal response of plasma facing components to produce synthetic tile surface temperatures within a strict cycle time limit of 8 ms.

This communication summarizes the motivation, requirements, design, and important advances to the operational and experimental understanding provided by these new tools.
Micro-/nano-characterization of the surface structures on the divertor tiles from JET ITER-Like Wall

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The study is focused on modification of surfaces of the tungsten-coated divertor tiles used in the first campaign (2011-2012) of the JET tokamak with the ITER-Like Wall (JET-ILW). The analyses by means of several material research techniques have been carried out at International Fusion Energy Research Centre (IFERC), JAEA Rokkasho. Samples, in the form of disks (17 mm in diameter), extracted from the inner divertor were studied: Tiles 1, 3 and 4, i.e. upper, vertical and horizontal targets, respectively. The selection of samples for detailed examination was based on the result of an imaging plate (IP) analysis. A specimen from the apron of Tile 1 was deposition-dominated. Thin and stratified mixed-material layers deposited on the original tungsten substrate were composed of Be, W, Ni, O and C. Their total thickness was ~200-400 nm. By means of transmission electron microscopy (TEM) large bubbles with size of over 100 nm were identified in that layer. They could be related to deuterium retention in the layer dominated by beryllium. Indeed, it has been reported earlier that the amount of deuterium on top of Tile 1 was highest among the three tiles. The surface microstructure of the sample from Tile 4 also showed deposition: a stratified mixed-material layer with the total thickness of ~200 nm. The electron diffraction pattern obtained with TEM indicated beryllium as the major component of the layer. No bubble-like structures have been identified. The surface of Tile 3, originally coated by Mo, was identified as the erosion zone. This is agreement with the fact that the strike point was often located on that tile during the plasma operation. In summary, the study revealed the micro- and nano-scale modification of the inner tile surface of the JET-ILW. Especially, complex mixed-material deposition layer could affect the hydrogen isotope retention and dust formation.
Formation of ammonia in N2 seeded discharges at AUG and JET

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10 See the Appendix of F. Romanelli et al., Proceedings of the 25th IAEA Fusion Energy Conference 2014, Saint Petersburg, Russian Federation

After the transition to full metal wall configurations at AUG and subsequently at JET, impurity seeding became necessary to maintain the divertor heat loads below material limits in H-mode discharges. Among the studied impurities, nitrogen (N) was found to be the most favourable option. However, it was also found that N2-seeding leads to formation of ammonia (NH3). Nitrogen and NH3 retained in surfaces can lead to increased tritium (T) retention. The presence of NH3 in the pump exhaust requires special arrangements for the operation of cryo-pumps and T recycling plant. Therefore, the quantification of NH3 production in N2-seeded discharges will also have direct implications on the design of the ITER tritium recycling plant. Past experiments at AUG and JET revealed a N-to-NH3 conversion fraction of 7 % and 2 % respectively, however the amount of detected NH3 in a series of identical dis-charges was found to continuously increase, suggesting that the steady state production at ITER with 100 s long pulses might be higher than expected from present data. The recent analysis of N2 seeded discharges at AUG and JET shows that the N2 seeding rate is the main discharge parameter determining NH3 production. However, NH3 also exhibits a significant legacy effect, which is visible as a gradually increasing level of detected NH3 in N2 seeded discharges, as well as an elevated level of released NH3 in subsequent non-seeded discharges. The release of retained NH3 was attributed both to plasma-wall interaction and flushing by gas injections, indicating that in-vessel surfaces can act as temporary pump for NH3, and that N2 seeding at ITER may lead to formation of a significant NH3 inventory in the vessel.
Optimization of the ASDEX Upgrade glow discharge system
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Glow discharge cleaning (GDC) and coating of the plasma facing components (PFC) is still crucial for fusion research machines to reach demands on plasma cleanliness for elaborate investigations. To correspond with latest experimental findings the GDC-system of ASDEX Upgrade (AUG) has been remodeled entirely. After transition to tungsten PFCs it becomes evident that Helium implanted during GDC contaminates subsequent plasma discharges significantly. This can be effectively minimized by alternately switch between short GDC pulses and longer purge phases. Therefore a dependable fast plasma ignition at operational pressure is required. Influenced by the need of space-saving design, enhanced reliability and simplified maintenance a radiation cooled tungsten-coated graphite anode was constructed based on the Wendelstein W7-X GDC electrode. Validated by extensive laboratory investigations a separate surface discharge spark device was developed. Four electrodes have been installed spread evenly around the circumference of the torus midplanes low field side each equipped with a starting device. To ensure high availability for use each single anode is supplied by its individual power supply unit. The starting devices are powered by individual capacities commonly fed via a high voltage power supply. This enables plasma starting times of less than 100 milliseconds and thereby also efficient pulsed GDCs with an on/off ratio of as less than 10/50 seconds. The paper gives an overview of the technical setup of the AUG GDC-system and summarizes the experiences of the first two operational campaigns.
Investigations on tungsten heavy alloys for use as plasma facing material

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Since 2014 ASDEX Upgrade (AUG) is using bulk tungsten tiles at the outer divertor strike-point. In two experimental campaigns more than 2000 plasma discharges with up to 10 s duration and 100 MJ plasma heating were successfully conducted, without impairment by the W tiles. However, an inspection after the campaigns revealed that a large number of tiles suffered from deep cracking, mostly attributed to brittle fracture. This could be overcome by alloying with rhenium which increases the ductility considerably or by the use of W fibre reinforcement which increases the toughness. However, these solutions either suffer from high costs or are not yet ready for routine use. An alternative solution – at least for non-nuclear fusion devices – could be the use of W heavy alloys as they are produced commercially by several companies. They consist of up to 97% W and Ni/Fe (or Ni/Cu) admixtures, they are readily machinable and considerably cheaper than pure tungsten. Their major drawbacks in view of the application in fusion experiments are the rather low melting temperature (1500 °C) of the Ni/Fe binder phase and their magnetic properties. In a first step W heavy alloys from two manufacturers were subjected to screening tests and cyclic loading in the high heat flux test facility GLADIS with up to 20 MW/m² and surface temperatures of up to 2000 °C, showing no macroscopic failure. SEM investigations show a segregation of Ni and Fe at the top surface after the thermal overloading, but no signs of micro-cracking. The longterm behaviour under plasma and electromagnetic load and the influence of the Ni/Fe preferential sputtering and the magnetic properties will be investigated with the AUG divertor manipulator. Moreover, the mechanical properties of the virgin and loaded material will be compared to those of tungsten used in the AUG divertor.
Microwave response of ITER diagnostic vacuum windows

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Diagnostic systems are essential for the development of ITER discharges and to reach the ITER goals. Many of these diagnostics require a line of sight to relay signals from the plasma to the diagnostic, typically located outside the torus wall. Such diagnostics then require vacuum windows that isolate the torus vacuum and crucially ensure tritium containment. While such windows are routine in many fusion experiments, ITER poses new challenges. The vacuum windows are Safety Important Components class 1 that must withstand all ITER loads. As a consequence, in many cases double windows are used. ITER is a long pulse machine with 20 MW microwave heating installed, giving rise to gradual heating of windows due to stray radiation. The particular microwave heating scheme at ITER may also - in case of an erroneous polarization setting - result in a highly focused beam that can be incident on a window and cause thermal shock. This paper looks at microwave aspects of ITER windows. The microwave response as a function of frequency is calculated for proposed arrangements. This response enables to assess the impact on diagnostic performance, such as the location in frequency space of the minima and maxima in reflection and transmission, as well as the attenuation of signal caused by absorption of the window. In the presence of microwave stray radiation, the absorption may lead to considerable dielectric heating of the window. Mitigation measures, such as reduction of the microwave power incident on the window and the application of coatings, are investigated. Such measures must be verified and qualified and dedicated measurements are discussed such as characterisation of dielectric materials in a low power resonator, directed beam measurements to verify reflection and absorption, high power tests to measure absorption and ITER vacuum qualification tests.
Investigation on feasibility of cleanliness assessment using test sample for ITER blanket shield block

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The ITER blanket shield block (SB) is one of the in-vessel components, which is designed to provide nuclear shielding and to supply the cooling water to vacuum vessel and external component. The ITER SB is classified the VQC 1A as vacuum classification and its manufacturing process and cleaning procedure shall comply with the ultra-high vacuum conditions necessary for machine operation and follow the requirements of the ITER Vacuum Handbook (VHB). And also the ITER SB shall be properly cleaned and dried before packaging and be ensured its cleanliness in accordance with ITER VHB. According to ITER VHB, cleanliness of vacuum components shall be verified with the wipe test and outgassing rate measurement. For wipe test, it is difficult to assess the quantitative cleanliness of vacuum components due to its subjective nature. Accordingly, the outgassing rate measurement may be needed for assessment of cleanliness for ITER SB. However, it is impossible to perform the outgassing rate measurement of ITER SB because additional activities, such as machining and any treatment, are prohibited after final cleaning process. Therefore, it needs to verify the satisfaction with cleanliness of ITER SB by using test samples. The objective of this study is to investigate the feasibility of cleanliness assessment by outgassing rate measurement using test samples for establishment of test procedure. The effect of cleaning method on the outgassing rate also investigated. The outgassing rate was measured at 100\degree{C} for 10hrs in accordance with ITER VHB.
Blanket manifold final design and validation

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The Final Design Review for the Blanket Manifold (BM) was successfully held in December 2015. Since the Conceptual Design Review, a concerted effort has been necessary on finalisation of the multi-pipe design, verification by analysis and practical validation to address challenging design requirements, and installation/maintenance processes. During normal operating conditions the BM provide pressurised cooling water to the plasma facing Blanket System with a nominal inlet pressure and temperature of 4.0MPa and 70°C, respectively. The design complies with RCC-MR 2007 Construction Rules for Mechanical Components of Nuclear Installations -class 2. The return circuits differ in temperatures by up to 40°C, which induce shear stress in neighbouring fixed pipe supports. The pipe to support interface has to accommodate conflicting requirements. On the one hand, it has to provide good thermal conductance in order to remove the neutronic heating. This would otherwise be conducted to the vacuum vessel (VV) attachment rails resulting in excessive thermal stresses. On the other hand, it has to be electrically insulated to mitigate excessive electromagnetic loads. For that purpose, a ceramic layer is used at each interface and VV attachment points. Thermal expansion of the multi-pipe bundles due to the average increase of outlet water temperature is accommodated by compliant support attachment legs. RAMI analysis results in the majority of the BM classified as RH class3, necessitating conceptual maintenance studies to be undertaken. Customised design features for both initial installation and subsequent remote maintenance have been introduced to accommodate the as built mounting points and profile of the VV and other in-vessel components. This paper provides details as to how the design has mitigated the conflicting requirements, the structural and EM verification of worst case operating scenarios, the practical validation to underpin design choices and the proven achievable build tolerances during the subassembly and installation processes.
Accuracy improvement studies for remote maintenance manipulators

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For ITER or the future DEMO remote maintenance system (WPRM), several types of special tailored automatic manipulators are needed for vacuum vessel (VV) component transportation, inspection, and removal from and replacement to the VV wall. These tailored manipulators, such as Multi-purpose Deployer, Articulated Inspection Arm (AIA), Diverter Cassette Mover etc., should be calibrated with very strict tolerance so as to handle the very large and heavy components going through the narrow gaps. The accuracy of automatic manipulator equipment depends not only on how accurate of its machining and assembling processes but also on how accurate of its control model can reflect its actual structures. For a remote maintenance manipulator, due to its specific design with big size, big payload and high environment temperature, both the static and dynamic error sources should be considered in order to improve the manipulator accuracy. In this paper, the accuracy improvement issues regarding to different robots are investigated, the static and dynamic error sources are taken into account for improving robot accuracy. The comparison results would answer the question of what is the relative importance of robotic static errors compared to dynamic errors. A 6-DOF (degrees of freedom) commercial industrial serial robot and a 10-DOF redundant hybrid ITER welding/cutting robot (IWR) at Lappeenranta University of Technology are used to carry out the corresponding simulation and experimental studies. This paper mainly focus on the general case studies, the results found in this research would be expected to form the basis of the proposed future connection with the work undertaken to support the ITER or the future DEMO remote handling systems.
Recovery from failures of ITER Blanket Remote Handling System

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How to recover from failures of components in radiation environment is an important issue of the ITER remote handling systems. Recovery operations of the remote handling systems must be performed remotely due to limitation of human access. For the ITER Blanket Remote Handling system, failure modes have been analysed, and the analysis has concluded that electrical failures of actuators, which are motors and resolver position-sensors, are need to be recovered and likely to happen. Those failures will be caused by insulation deterioration by radiation and failures of electrical connectors. To make the ITER Blanket Remote handling System recoverable from those failures, we developed two recovery methods, and this paper presents them. The first method is rescue operation using manipulators. The failed actuator can be driven by external force because it is not damaged mechanically in the considered situation. We designed a rescue tool to drive the failed actuator externally and rescue manipulators to handle it. In addition, we studied reach of the rescue manipulators and concluded that the ITER Blanket Remote Handling System can be recoverable using the rescue tool and manipulators. The second method is applicable for resolver failures: connecting another resolver (hereafter called ‘dummy resolver’) to the motor driver instead of the failed resolver. Then the driver regards position of the dummy resolver as the current position of the motor. By controlling the driver to keep ‘the current position’, the driver tries to keep the motor position same as dummy resolver. Thus, the motor with the failed resolver can be driven by rotating the dummy resolver. We tested and confirmed that the motor follows rotation of the dummy resolver under load up to the rated torque by the rated rotation speed, and concluded this method is applicable to the ITER Blanket Remote Handling System.
Development of In-Vessel Pipe Welding Tool for ITER Blanket Remote Maintenance

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The ITER blanket module has hydraulic connections to the cooling water manifold. The connections are designed to be cut and re-welded remotely in the vacuum vessel during blanket maintenance due to irradiation of in-vessel components after D-T experiment. In course of the R&D activities for in-vessel pipe welding, a study [1] demonstrated that good weld quality can be achieved by correcting misalignment of weld groove, in parallel with optimizing welding conditions such as heat input and gas pressure. Thus correcting pipe misalignment and confirming that the misalignment is corrected are critical for reliable in-vessel welding of ITER hydraulic connections. This paper presents development of in-vessel welding tool for ITER blanket remote maintenance, which consists of pipe misalignment correction tool, pipe misalignment measurement tool and welding tool. Whole concept of tooling operation by tool manipulator attached to the ITER in-vessel transporter collaborating with a dexterous manipulator for cable handling is also presented. In particular, test results of a pipe misalignment correction tool and two different types of pipe misalignment measurement tools using laser light are shown. The misalignment measurement tools are those by (a) reflection intensity measurement of incident laser to the pipe weld groove by photodetector, and (b) image observation of weld groove by near-infrared laser, respectively. In both methods, in view of limited accessibility to the hydraulic connections in ITER blanket modules, measurement of weld groove was done from the inner side of φ42 mm inner diameter pipe with 3 mm thickness. Moreover, these tools are designed to be integrated with laser welding tool. Integrated testing of pipe misalignment correction, weld groove measurement and welding is to be carried out.
Development of a prototype work-cell for validation of ITER remote handling control system standards

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An integrated control system architecture has been defined for the implementation of ITER Remote Handling (RH) equipment systems. The RH Core System (RHCS) is a standard software platform used for the development of ITER RH equipment controller applications to facilitate the integration with this system. It installs on top of the CODAC core system and provides a uniform platform for the development of remote handling control applications. The RH core system is packaged using several open source tools and libraries including OROCOS, KDL, GSL, Eigen, Eclipse, maven, etc. It emphasizes the usage of OROCOS real-time tool chains in building individual control system components that are highly configurable and interactive. The communication between the control room and the embedded control applications is achieved using the standard Controller Interface Protocol defined by ITER. Prototyping work has been carried out for the development of individual sub-systems including RH equipment controller, viewing system, virtual reality monitoring system and RH plant controller built on the RH core system using the standard network communication protocols of ITER. All the individual sub-systems have been integrated into a prototype work cell and successfully tested on a COTS robot manipulator to study the real-time performance aspects of the software ecosystem. This paper presents the design & implementation of the prototype work-cell and concludes by suggesting recommendations for the next version of the ITER RH core system.
ITER Hot Cell – Remote Handling System maintenance overview
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ITER is a large scale fusion device designed to study the high temperature fusion reaction between tritium and deuterium. The success of a tokamak-type fusion reactor will depend to a great extent on developing reliable and safe methods of carrying out routine maintenance and repairs remotely. Remote Handling System (RHS) are used to perform remotely the maintenance of the vacuum vessel. They will be contaminated during exposure inside ITER Vacuum Vessel with activated dust, tritium and beryllium. RHS will be transported to the Hot Cell Facility at level B1/B2 to be remotely decontaminated before hands-on maintenance operation. Maintenance is made of repair, changing parts, testing and re-commissioning activities. The first commissioning and the training on RHS will take place in a cold Test Facility, free from nuclear or Beryllium hazard. In summary, The system for RH Equipment maintenance shall provide the means to: Remote decontaminate the Remote Handling Systems. Support hands-on maintenance tasks of the Remote Handling Systems. Support recommissioning tasks of the Remote Handling Systems. Support rehearsal tasks (checking & training) of the Remote Handling Systems. In this paper the results of concept design engineering investigation of the routine remote maintenance and repairs is developed. Maintenance requirements for a nuclear fusion facility, are rather similar to those of JET, but magnified. A dedicated facility of a unique scale is planned to support ITER remote maintenance activities. The paper focuses the proposed concept for process and equipment of the hot cell with consideration on integration and safety matters.
Irradiation tests of radiation hard components for ITER blanket remote handling system

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The ITER Blanket Remote Handling System (BRHS) will handle the blanket modules (BMs), which can weigh up to 4.5 ton and be larger than 1.5 m, stably and with a high degree of positioning accuracy. When the ITER has stopped plasma operations for maintenance, the BRHS will be installed in the vacuum vessel, whose components are radioactive, to remove and install the BMs. Therefore, the BRHS will be operated in a high radiation environment (up to 250 Gy/h) having an estimated total dose of 5 MGy during maintenance (maximum of two years). The radiation hardness requirement for ITER BRHS components is 1 MGy total dose. As components may degrade by gamma irradiation, some equipment is expected to malfunction which causes delays in the in-vessel maintenance schedule. Therefore, failure mode and effects analyses (FMEA) were performed on the BRHS components and FMEA results suggest trouble with the power supply and signals due to degradation of the insulation of electrical components, and malfunctioning motors due to degradation of the lubricants of mechanical components. Material selections and irradiation tests were performed to AC servo motor and non-halogen cable in the past study. As a result, radiation hardness of 8 MGy and 3 MGy were confirmed for AC servo motor and non-halogen cable, respectively. In this study, as the polymer material for the O-rings, bellows, cable sheaths, and coating materials is of the utmost priority in the design of the BRHS, additional material property tests to verify radiation hardness were performed after the candidate materials were irradiated with gamma rays up to 5 MGy. After selection the radiation hard materials for BRHS by property test, function test such as sealing tests and repeat test were performed as the second step to confirm that the materials could still function properly as an O-ring and bellows.
A hybrid DE and PSO algorithm for numerical solution of remote maintenance manipulators

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In the ITER or the future DEMO reactor systems, 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 automatic manipulators. The structure of these manipulators can be designed as a pure serial articulated arm or a pure parallel mechanism, but for the sophisticated remote handling tasks requiring large workspace and high payload/weight ratio, it would be designed as a hybrid structure combined by these two mechanisms. In this paper, a global optimization method combined by differential evolution (DE) and modified particle swarm optimization (MPSO), here we call it as DEMPSO, is developed to obtain the numerical solutions of robot kinematics. DEMPSO algorithm combines the advantages of the global optimization of differential evolution (DE) and the fast convergent rate of particle swarm optimization (PSO). The general case studies focus on the numerical solutions of the inverse kinematics of a 6-DOF serial robot, the forward kinematics of a 6-DOF parallel robot, and the forward kinematics of a 10-DOF hybrid redundant serial-parallel robot designed for ITER vacuum vessel remote maintenance. The comparison study of other optimization methods including ant colony optimization (ACO) algorithm, differential evolution (DE) algorithm, particle swarm optimization (PSO) and modified particle swarm optimization (MPSO) algorithm have also been investigated to validate the robust and efficiency of the proposed DEMPSO algorithm. The results found in this paper would be extrapolated to solve the kinematic problems of the ITER or the future DEMO remote handling manipulators. The proposed optimization algorithm can also be used for the static and/or dynamic parameter identification in robot calibration systems.
Assessment of unmanned aerial vehicles for reactor inspection

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Experimental fusion reactors aim at the exploration of the nuclear fusion as a viable energy resource. Remote Handling Systems (RHS) are specially designed for regular operations of inspection and maintenance inside the reactors, such as the In-Vessel Transporter, an extendable robotic arm deployed in the equatorial level of ITER. The reactor is shutdown during the installation and operation of the RHS, which is time-consuming and expensive. Unmanned Aerial Vehicles (UAV) are able to perform simple inspection missions inside the reactor before the RHS operations. The information acquired by the UAV through the reactor provides the ability to previously setup the inspection and maintenance operations, yielding time and cost benefits. This work presents an assessment of common UAV (e.g. quadcopters), to perform simple inspection missions inside the reactor during its shutdown. Such UAV are able to transport different on-board sensors to get an insight view of the blankets and other elements inside the reactor, while providing the maneuverability and endurance to perform the inspection missions. The costs of producing, maintaining and operating UAV are reduced when compared to the time and costs that can be reduced with the valuable information acquired during the flight. To achieve the maximum flexibility for different type of missions and also for the replace ability following radiation exposure, the UAV is composed of modules, such as frame, propellers, motors, sensors, main control unit and batteries. This work also proposes a multi-criteria optimization approach to find the best UAV design and configuration to operate inside the reactors, taking into account the time duration of the mission, the required maneuverability, the endurance to rad-hard conditions and replace ability of the modules, the total weight, and the total cost of the UAV. A set of optimized solutions are presented and compared to the performance of different commercial off-the-shelf solutions.
Robust grasping motion control with force feedback for EAMA robot in fusion tokamak application

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The EAMA robot is a long slender arm for tokamak inspection and maintenance. In such conditions, grasp techniques ignoring or trying to avoid contact with the components of the vacuum chamber brings bottlenecks on the system control. During the grasping and releasing objects the contact with vacuum chamber is a critical condition for providing robust and achievable solutions of robot control. In this paper, a kinematic model for motion coordination and control of EAMA system is derived, which calculate the object pose from the joint variables of each link from a suitable set of contact variables. This interactive, compliant primitive grasping is reached by a series of position-force combined pre-grasping, landing and post-grasping strategy of EAMA placement. This model is adopted to design a control scheme to achieve a desired target motion and keep desired contact forces applied to the object in order to preserve safety. The results demonstrate effectiveness and robustness of simulation study case for the proposed application.
P4.137

Operators’ accessibility studies using virtual reality

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The development of fusion plants is more and more challenging. Compared to previous fusion experimental devices, integration constraints, maintenance and safety requirements are key parameters in the ITER project. Components are designed in parallel and we must consider integration, assembly and maintenance issues, which might have an impact on the overall design. That also implies to consider the feasibility of installation and maintenance of elements by operators. Virtual reality (VR) provides tools to optimize such integration. In 2010, the CEA IRFM decided to take the gamble of using VR tools during the life cycle (from design to operation) of a fusion component. The VR platform is intensively used in the design and assembly studies of WEST components. In particular, the assessment of the interventions of the operator is part of the process. To study this aspect, the use of static manikins is quite frequent in the industry. However, more complex studies like the feasibility of assembly and maintenance tasks in complex and very confined environments require enhanced features such as dynamic and biomechanically realistic virtual humans. We also study the contribution of tactile feedback to improve physical presence and interaction in the virtual environment (VE), which is very important for the validation of feasibility and the ergonomic of posture and gesture of the operator involves. In particular, we showed that a better behavior regarding physical element of the VE can be obtained with both the presence of a dynamic representation of the subject’s body and a tactile feedback. In this paper, we will present integration studies involving operators and recent advances in the assessment of maintenance feasibility.
Mascot is a two-armed dexterous master-slave telemanipulator device linked by force-reflecting servomechanisms, giving the operator a tactile sensation of doing the work. Mascot version 4.5 is currently in use at the Joint European Torus (JET) experimental nuclear fusion facility. Its role is to maintain the inside of the reactor vessel without the need for manned entry. The slave is typically attached to a boom which transports it to the work area. The Mascot 6 project, funded by EFDA, was initiated to address reliability and availability issues arising as a result of obsolete technologies. In particular, the Mascot actuators based around obsolete 2-phase AC induction motors are to be replaced with actuators based on commercial off-the-shelf (COTS) Permanent Magnet Synchronous Motors (PMSMs). As a consequence of its highly integrated, monolithic design, the entire Mascot control system, including servoamplifiers, controllers, control software and HMI needs to be redesigned. The Mascot6 control system is designed to maximise Reliability, Availability, Maintainability, and Inspectability (RAMI) of the system, as well as providing significant future-proofing. Standard interfaces are used to integrate subsystems wherever possible in order to maximise future proofing, and capacity for future modifications. Mascot 6 control system utilises the advanced RACE generic framework for interoperable control. Advanced Computer-Aided-Teleoperation features such as dynamic force compensation and load cancellation are presented. A master safety system has been developed in order to maximise safety of the operator, whilst working with the master device as a collaborative robot. Significant trials and development have also been conducted as part of the Mascot 6 project into design features and control techniques to meet the demanding performance requirements of the system, including producing a smooth torque output at or near stall, which is an unusual requirement for many motor and drive manufacturers, but is vital for teleoperation with transparent haptic feedback.
P4.140

Application of Ultrasonic technology in CFETR vacuum vessel R&D

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Full penetration welding and 100\% volumetric examination of weld joints are strictly required for all welds of pressure retaining parts of the CFETR Vacuum Vessel (VV) according to the design manual. However not every welding joint can be tested using RT method due to component structure and welding position. Therefore, the ultrasonic testing (UT) has been selected as an alternative method. Considering the misjudgment and undetectable of defects in the austenitic stainless steel welding by applying the traditional ultrasonic testing method, the most advanced phase-array ultrasonic technology (PAUT) has been chosen for our testing procedure design. By using the ultrasonic simulation, the most adapted inspection method is determined at the first time. Applying of phased array dynamic focusing technique shows its advantage in this application. The precision of the defect position and the signal/noise(S/N) have been improved comparing the conventional UT method. This work shows that the PAUT technique has excellent detectability and applicability for the austenitic stainless steel weld in the CFETR VV.
Research of NG-TIG welding technology on full-scale sector prototype of CFETR vacuum vessel

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With the development of CFETR engineering design, a full-scale sector prototype of vacuum vessel has been carried out as one of the major R&D projects. The welding structure between vacuum vessel sectors in field assembly is modeled in this prototype, and NG-TIG is taken for an applicable welding strategy with small welding deformation, high-quality welds and excellent adaptability to the structure. All-position welding process test including flat position, vertical position and over-head position are launched, then relevant test results are obtained including non-destructive testing, mechanics performance testing, macro and micro weld detection, permeability testing, and the optimal process parameters. All-position welding system solutions are given in this paper, and finite element analysis by SYSWELD is executed and the results are compared with the actual measurements.
A Study on Assembly Technology of the CFETR 1/32 Vacuum Vessel

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Chinese Fusion Engineering Testing Reactor (CFETR) is a superconducting magnet Tokamak, it has the equivalent scale with complementary function to International Thermonuclear Experimental Reactor (ITER). The vacuum vessel (VV) which has a double-layer structure will remove the heat generated during operation. The VV will provide a high-vacuum environment for the plasma, improves radiation shielding and plasma stability, and provides support for in-vessel components. The CFETR VV is composed of 16 sectors, the angle of each sectors is 22.5°. The Research and Development (R&D) of the key technologies to the VV manufacture have been carried out a few years ago by Institute of Plasma Physics Chinese Academy of Science (ASIPP), including Narrow-Gap welding, cutting and non-destructive testing (NDT), Poloidal Segment (PS) and sector assembly technologies, etc. ASIPP is constructing a 1/8 sector real size VV now. The manufacture of the PS for first 1/32 sector VV have been completed in 2015. The PS will be assembled into a whole this year. This paper will describe the study of the assembly technology for the CFETR 1/32 Vacuum Vessel.
R&D activities of tritium technology on the broader approach in phase 2

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In the one of Broader Approach (BA) activities aiming to the development for a DEMO fusion reactor, the R&D of tritium technology has been carried from 2007. The period consists of Phase 1 (2007-2010) and Phase 2 (2010-2016). International Fusion Energy Research Center (IFERC) including DEMO R&D building was constructed in Rokkasho BA site of Japan. The R&D building is a facility to handle tritium, other radioisotopes and Beryllium. The R&D of tritium technology carried out by not only JAEA but also collaborative R&D studies with Japanese universities. Presentation gives the summary of R&D activities on tritium technology in Phase 2 together with overlook of significant results.
Hydrogen inventory control for vanadium alloy by metal powder mixing in molten salt

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One of the major concerns for molten salt breeding blanket system is the low tritium solubility, high equilibrium tritium pressure in other words, of the molten salts including FLiBe, FLiNaBe and FLiNaK. Owing to this, vanadium alloy (V-4Cr-4Ti) has been thought to be inappropriate as a structure material in molten salt breeding blanket because of its high tritium solubility. The concept of hydrogen absorbing metal powder mixture for molten salt proposed by Sagara (2014) is the promising method for this issue. The authors (2015) has already experimentally confirmed that titanium powder successfully increased the effective hydrogen solubility of molten salt FLiNaK in more than 5 orders and will suppress the hydrogen concentration in structure materials. In this work, pure vanadium and vanadium alloy plates are immersed in titanium mixed FLiNaK stored in Ni crucible and hydrogen gas is supplied. Hydrogen inventory in the alloy is quantified by TDS followed by the cooling and removing solid FLiNaK. The effect of metal powder on the hydrogen transfer from gas phase to the vanadium plate through molten salt mixture is investigated to be presented in the conference.
Numerical simulation of purge gas flow in binary pebble beds based on DEM-CFD method

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The neutron multiplier and the tritium breeder materials are made into millimeter-sized particles and arranged in the solid breeder blanket. Helium (mixed with 0.1\% content of H\textsubscript{2}) is used as the purge gas to sweep tritium out when it flows through the pebble beds. Previous research shows that binary pebble beds present a better performance in tritium breeding than unitary pebble beds. Since the flow characteristics of helium in binary pebble beds are very important parameters for the evaluation of tritium sweep capability and the design of tritium recovery system, DEM-CFD method was induced in this paper to determine the flow characteristics. The distinct element method (DEM) was applied to produce a geometric topology of binary pebble beds by directly simulating the contact state of each individual particle using basic interaction laws. Based on the geometric topology, a computational fluid dynamics (CFD) model was built to analyze the flow characteristics including pressure drop, velocity field and so on. In the current study, pebble beds with different diameter ratios and different packing factors were simulated. It was found that the pressure drop of binary pebble bed increased greatly with the rise of the packing factor and was much larger than mono-sized pebble bed. Besides, strong nonuniformity of velocity distribution was observed from the velocity field and stagnation flow was more obvious than single-size pebble bed. This method can be well used to optimize the particle diameter ratio of a preferable binary pebble bed in order to obtain appropriate pressure drop and acceptable tritium sweep capability for the design of fusion blanket.
Analysis of low pressure hydrogen separation from fusion exhaust gases by means of superpermeability

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The effect of superpermeability is capable of separating hydrogen and its isotopes out of gas mixtures at low pressures even against a pressure gradient. This process allows strongly enhanced permeation. It relies on metal membranes that are exposed to atomic hydrogen. If the surface inhibits the chemisorption on its surface, the atomic hydrogen can still enter the bulk, but hydrogen recombination on the surface is suppressed. Only few molecules are created that can leave back into the gas phase. The concentration gradient of hydrogen drives the diffusion through the membrane. This is an ideal concept for the implementation of Direct Internal Recycling (DIR) in future fusion machines. DIR is one of the measures aiming to drastically decrease the tritium inventory of a Demonstration fusion power plant which will be necessary due to tritium availability and regulatory issues. In the HERMES facility at KIT superpermeability is investigated. In this publication permeation measurements are shown and interpreted. During the measurements a change of the surface properties was found. These stability issues are discussed in detail. A theoretical model to describe this effect is outlined and benchmarked against the experimental results.
On the study of catalytic membrane reactor for water detritiation: effect of reactions kinetics

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In the framework of tritium recovery from tritiated water, efficiency of packed bed membrane reactors have been successfully demonstrated. Thanks to protium isotope swamping, tritium can be recovered from tritiated water under the valuable Q2 form (Q = H, D or T) by means of isotope exchange reactions occurring on catalyst surface. The use of permselective Pd-based membrane allows withdrawal of reactions products all along the reactor, and thus limits reverse reaction rate to the benefit of the direct one (shift effect). The reactions kinetics, which are still little known or unknown, are generally assumed to be largely greater than the permeation ones so that thermodynamic equilibriums of isotope exchange reactions are generally assumed. In this paper, the influence of reaction kinetics is evaluated thanks to experimental approach. A dedicated fixed bed reactor filled with catalyst was used to assess the deviation to theoretical thermodynamic equilibriums under different operating conditions corresponding of the range of operating conditions foreseen for the catalytic membrane reactor. Effects of temperature, inlet gas composition and residence time were studied. This led to the simplification of the reaction scheme as for all operating conditions tested, thermodynamic equilibrium could be demonstrated.
The coolant purification system in DEMO: candidate technologies, requirements and interfaces

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The blanket concepts investigated under the EUROfusion program rely on water or helium as the primary coolant medium; the main duty of the coolant is to recover the thermal power from the first wall and the blanket units and drive it into the Primary Heat Transfer System (PHTS). The coolant path goes through three different systems: the breeder, the tritium plant and the PHTS. In the breeding region, due to the high temperature and the reduced thickness of the coolant pipes, some of the produced tritium can permeate into the coolant loop; therefore the main duties of the Coolant Purification System (CPS), located inside the tritium plant, are the extraction of tritium and the control of the coolant chemistry. There are several reasons why it is important to remove tritium from the coolant: 1) to avoid the release of tritium via permeation and/or coolant leakages into the working environment, 2) to keep the release of tritium into the environment lower than the allowable limit (tritium present in the primary coolant can permeate into the secondary coolant and thus reach the environment), 3) to control the tritium balance that is essential for monitoring the blanket performance during the different operational phases. The paper provides a description of the principal candidate technologies to be used inside the CPS both in case of helium and water by considering the very challenging CPS scenario characterized by a low tritium amount diluted in a huge flow of coolant. It also identifies the main requirements (i.e. fraction of coolant inside the CPS, tritium concentration in the coolant, CPS efficiency, etc.) and interfaces of the CPS.
Fusion plasma exhaust is generally composed of unburned fuel (deuterium and tritium), helium and few impurities. However for a metal wall machine (like DEMO) that reaches elevated powers, a certain amount of plasma enhancement gas (nitrogen, Ar, Ne, etc.) could be used as seeding for enhancing the radiative power and decreasing the power load over the plasma facing components. The recovery of these Plasma Enhancement Gases (PEG) could be beneficial because of the high flow rates required, and to limit the load placed upon the exhaust detritiation system. In this work, the application of ceramic porous membranes for the separation of PEG from other plasma exhaust gases is studied. The gas permeability through porous media of hydrogen, helium and a number of inert gases of potential interest (N2, Ne, Ar, Kr, Xe) has been assessed via the models of Knudsen and Poiseuille. A parametric analysis taking into account the effect of temperature (20 and 300 °C), pressure (100 kPa and 1 MPa) and pore size of the membranes (0.1 nm, 10 nm, and 1 mm) has been undertaken to evaluate the capability of porous membrane systems to recover PEG from the exhaust gas in terms of separation factors. The preliminary design of a membrane module is also carried out.
Heavy water decontamination tests through a Pd-membrane reactor

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Pd-based membrane reactors are well-known technologies in the fuel cycle of the next fusion plants. In this work the application of Pd-Ag membranes have been studied in order to recover tritium in both molecular (Q2) and, especially, oxidised (Q2O) form in the tritium extraction system (TES) of the HCPB blanket. The membrane reactor is made up of a Pd-Ag membrane tube filled with a catalyst. It allows to operate the permeation and chemical reaction in the same device as well as to reach higher reaction conversion and decontamination efficiency. Other advantages of such membrane modules are related to their high performances in terms of hydrogen permeability and selectivity, to their modularity (easy to scale up) and to their continuous operation (regeneration not required). This kind of reactor is proposed for tritiated water decontamination through two types of reactions: Isotopic Swamping (IS) and Water Gas Shift (WGS). In order to minimize the tritiated by-products (especially methane) and, at the same time, to promote the WGS or IS reactions, this work evaluates the performances of the membrane reactor for the two reactions (WGS and IS) under different operating conditions (temperature, pressure, concentration) by testing different catalysts. The tests results are reported in terms of decontamination factor (the amount of gaseous deuterium recovered from heavy water) by comparing the effectiveness of different catalysts. Especially, the capability of a new catalyst to minimize the formation of tritiated methane (by-product of the WGS) is tested.
Perspectives of the Vacuum Sieve Tray method to extract tritium from Pb-16Li at TLK

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Tritium self-sufficiency and management in nuclear fusion power plants is still challenging. Advanced technologies to extract tritium from lead lithium (Pb-16Li) as possible breeder material are required. The Vacuum Sieve Tray (VST) method consists in pushing Pb-16Li through a tray of submillimeter scaled nozzles towards a chamber maintained under dynamic vacuum. At the exit of each nozzle, an instable liquid jet is formed which breaks up in droplets, whose oscillations (up to 200 Hz) are presumed to enhance the hydrogens mass transport to their surface. The VST has experimentally proved to be efficient to extract deuterium, used to mimic tritium but avoiding the constraint of radioactivity [1]. We first developed a model to calculate the extraction efficiency, describing in details the pressure losses along the Pb-16Li flow. It was applied to design a single-nozzle VST experiment operable with tritium. A multi-nozzle VST experiment with deuterium was also developed to mitigate the risks and to tackle technical issues (Pb-16Li solidification, corrosion rates, Sieverts’ constant...). Moreover, this experiment also extends the possibilities of the single-nozzle concept, as it includes multi-nozzle trays, where deuterium extracted from one droplet may be reabsorbed by another one limiting the extraction efficiency. This contribution presents the complementary goals and expectations of the single-nozzle and multi-nozzle experiments to be performed respectively with T2 and D2. Experiments intended to validate and adjust the models used to calculate extraction efficiency and to quantify multi-nozzle disturbances are proposed and discussed. The confrontation of the theoretical approach and experimental results will allow understanding the impact of the geometry (nozzle, tray and set-up) on the extraction efficiency depending on the mass flow rate. [1] F. Okino, K. Noborio, R. Kasada, S. Konishi, Fusion Sci. Technol., 64 (3), 543-548, 2013.
Effect of halogenated gas on detritiation efficiency of the detritiation system

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Effect of halogenated gas on detritiation efficiency of the detritiation system was investigated. In order to accelerate tritium safety of the Japanese DEMO reactor, the detritiation system should be designed taking possible off normal events such as fire carefully into consideration. In an event of fire in a tritium processing room, halogenated gases such as hydrogen chloride, halogenated hydrocarbons and chlorine would be produced from the burning of the electric cables which insulation is made from polyvinyl chloride. In the presence of these gases, the detritiation system which key components are the catalytic reactor for the oxidation of tritium in combination with following tritiated vapor absorber should not lose their conversion and detritiation efficiencies. Concerning the activity of platinum catalyst for the oxidation of tritium, we evaluated the decrease in activity of platinum catalyst in the presence of halogenated gases. In order to avoid the steep decrease in activity, a noble catalyst alloyed with platinum and palladium showed an outstanding proof against halogenated gases. Turning to the effect of tritiated reactions in the catalytic reactor on conversion efficiency, formation of tritium chloride by the reaction between tritium and chlorine had an impact on conversion efficiency at temperature of catalyst below 373 K. As for water absorber, a molecular sieve moderately decreased its water absorbing capacity especially in the presence of chlorine. The results of this study reveal that the selection of catalyst is the key to preserve the detritiation efficiency of the detritiation system in the presence of halogenated gases.
Hydrogen isotope delivery performance of a DU hydride bed under various preheating scenarios

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The hydrogen isotope storage and delivery system (SDS) is a complex system that includes many individual components. One of the most important parts of the SDS is a metal hydride bed, which stores and delivers the hydrogen isotopes and pure gases required for a nuclear fusion reactor. We have been developing a metal hydride bed using depleted uranium (DU). The hydrogen delivery performance of the metal hydride bed is an important part of satisfying the fueling requirements in accordance with the fusion fuel cycle scenario. Because DU hydride can be heated to the very high temperatures at which it is possible to pump hydrogen isotopes, we designed and fabricated a mock-up DU hydride bed without any hydrogen delivery pump. However, the application of preheating scenarios seems to be essential for more rapid and reliable hydrogen delivery performance under pressure and temperature constraints. In this study, the effect of the preheating scenarios was experimentally investigated using the DU hydride bed with hydrogen gas. The application of preheating scenarios was found to improve the hydrogen delivery performance of the DU hydride bed. In addition, two empirical equations based on our experimental results were suggested to predict the effect. One equation is mainly composed of two terms to consider the inner pressure of the DU bed and the hydrogen atomic ratio in DU hydride. The equation is used to determine Pressure-Composition-Temperature (PCT) curves. The other equation is mainly composed of a correction factor, a Boltzmann factor, and two terms to consider the inner pressure of the DU bed and the hydrogen atomic ratio in DU hydride. The equation is used to predict the variation of the hydrogen atomic ratio in DU hydride with the lapse of time. The numerical simulation results by the equations had a good agreement with the experimental results.
The hydrogen isotope storage and delivery system (SDS) is a part of a nuclear fuel cycle. It is a complex system that is composed of numerous components such as a metal hydride bed, measuring tank, and other essential components. Depleted uranium (DU) was chosen as a hydrogen isotope storage material because of its rapid reactivity. We designed and manufactured the DU hydride bed to store the hydrogen isotopes and supply them to the nuclear fusion reactor. The hydrogen recovery performance of the bed was evaluated through both experimental and numerical investigations to determine the best recovery environment. In this study, we consider the effect of the inner temperature of the DU hydride bed on the hydrogen recovery performance. The inner temperature of the DU hydride bed was controlled using a proportional–integral–derivative (PID) controller to experimentally determine the overall effect. Two empirical equations for the pressure-composition-temperature (PCT) curve and the for hydrogen atomic ratio in DU hydride were reformulated and were used to predict the effect numerically.
Dynamic simulation of a multicomponent distillation column for D-T separation

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Cryogenic distillation (CD) process is being employed, among other applications, in tritium separation technologies and in case of ITER is one of the key processes in the fuel cycle. The ITER Isotope Separation System has to process by cryogenic distillation various mixtures of H-D-T depending from the various torus operation scenarios. Cryogenic distillation has also been employed to separate and concentrate tritium in a CANDU water detritiation system. Dynamic simulation of a distillation column gives information on the behavior of the system when fluctuations in flow feed or feed concentration may occur, with direct impact on the design of the control system. The objective of this work is to present a mathematical model for dynamic simulation of a multicomponent distillation column for D-T separation. The procedure of dynamic simulation is based on Lewis – Metheson method and tridiagonal matrix method for design, respectively simulate a multicomponent distillation column; the model is used to determinate the time required to reach steady state into the entire distillation column, after the occurrence of a process perturbation (e.g. modification of a feed flow, feed concentrations).
P4.156

Investigation of some critical scenarios due to various failures of ICSI Cryogenic Distillation System

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During normal operation of a CANDU reactor, large amounts of tritiated heavy water is being produced as result of neutron absorption by the heavy water used as moderator and cooling agent. Tritium in the heavy water, being radioactive, brings a significant contribution to the personal doses and also represents an environmental hazard if a waterspill occurs. The Pilot Plant for T2 and D2 separation from Rm. Valcea (Romania) is an experimental facility built to confirm the technological data and the functional characteristics of equipments for designning and construction of an industrial facility capable to process the tritiated water from Units 1 and 2 from NPP Cernavoda. The technology employed is based on Liquid Phase Catalytic Exchange (LPCE) and Cryogenic Distillation (CD) processes and the whole system is in preoperation stage. The CD system consists of a cascade of four distillation columns placed inside a vaccum insulated coldbox and a refrigeration unit which provides the cooling capacity for the columns condensers. This paper analyzes some critical scenarios due to single failure mode or a combination of failures such as loss of cooling capacity, loss of cooling agent into the columns, and loss of electric power (loss of cooling power and vacuum system shut down), cases seen as conservatives for the system in order to investigate the behaviour during abnormal operation or accident. The study presented in the paper is concerning the Pilot Plant for T2 and D2 separation from Rm. Valcea but gives information and some references for the methodology that can be implemented for CD Sytem from NPP Cernavoda and ITER Isotope Separation System.
Gas distribution system manifold design for ITER gas injection system

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Being part of the ITER fuelling system, the primary functions of the Gas Injection System (GIS) include providing gases for plasma discharge, wall conditioning, and neutral beam injectors. The Gas Distribution System (GDS) is a key sub-system of the GIS, which shall distribute gases obtained from the Tritium Plant, to the Gas Valve Boxes for the Pellet Injection System, Gas Fuelling System, Distruption Mitigation System and Neutral Beam system, via a distribution manifold. This paper introduces an overview of GIS and focuses on the engineering design of the GDS, especially the dedicated manifold. The main content includes: (1) Design requirements; (2) Design features; (3) Code (4) Components classifications; (5) Boundaries and interfaces; (6) Component design; (7) On site assembly plan. Structural analysis of GDS has been carried out by the FE method based on the load specification, whose results endorse the safety of the current GDS in given conditions. RAMI (Reliability, Availability, Maintainability & Inspectability) analysis has been performed; meanwhile the tritium safety and accident/incident measures are considered. Besides, the manufacturing feasibility has been verified by the test components. All the current results introduced in this paper show the capability of GDS Manifold to achieve the expected functions and the compatibility with the latest space reservation. Final design review of GDS Manifold has been held at the domestic agency (CNDA) based on the ITER design procedure.
State of the art and perspective of high-speed pellet injection technology

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The injection of cryogenic pellets from the low field side (LFS) has long been in use for core fueling of fusion devices. However, with higher plasma temperatures and bigger sizes, this technique becomes increasingly inadequate to ensure effective core particle deposition; injection from the high field side (HFS) has shown better results, despite the severe limitations imposed to the pellet speed ($\leq 300$ m/s) by inboard accessibility. For future burning plasma reactors, an alternative approach may be that of injecting high-speed pellets from the HFS, through suitable “free-flight” paths, eliminating curved transfer systems [1]. Furthermore, the expected length of the plasma discharges will require steady-state repetitive systems, capable of firing pellets at frequency no less than 10 Hz. ORNL and ENEA have been collaborating on high-speed injectors since 1990; they successfully realized a high-speed repeating pellet injector (2.55 km/s at 1 Hz), by combining an existing ORNL D2 piston extruder, and an ENEA two-stage gun [2]. Since then, good progress has been achieved on both fronts of steady-state extruders [3], and operation and reliability of two-stage guns [4]. A comprehensive R&D program is therefore proposed, including several innovative techniques, to investigate how far speed limits and repetition rates of combined two-stage guns and steady-state extruders technologies can be extended. Simulation results are presented to determine optimized pellet injection locations on the basis of the expected plasma parameters for future devices, either under construction such as JT60SA or being proposed.

[1] A. Frattolillo et al., this Conference
P4.159

Pellet injectors for EAST and KSTAR tokamaks

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High frequency pellet injectors have been developed for edge localized mode mitigation and plasma fuelling of the EAST and KSTAR tokamaks. Each pellet injector is able to inject solid deuterium or hydrogen pellets at steady state mode. Both injectors consist of a continuous ice generator based on a screw extruder cooled by liquid helium and pneumatic punches for pellet fabrication and acceleration. The EAST pellet injector is able to inject 1.5 mm diameter and 1.2-1.8 mm length pellets at frequency 1-50 Hz with velocities up to 230 m/s. Injection reliability over 95% has been confirmed during several 100 s cycles of continuous D₂ pellet injection at 50 Hz. Pellets are injected by two modules working in turn at 1-25Hz each. Injection modules are placed in one vacuum chamber and cooled by a common cooling circuit. Pellets injected by each module fly towards a tokamak chamber through a common guide tube. Two valves and an electromagnet are used for propellant gas admission to drive a puncher forward-backward inside the ice generator to form a pellet, accelerate it and remove gas from each module. The KSTAR pellet injector has been designed to inject 2 mm diameter and 1.5-2.0 mm length pellets with minimal velocity 200 m/s at frequency up to 20 Hz. Contrary to the pellet injector for the EAST tokamak, pellets for KSTAR are formed by a puncher driven by a valve at pressure which can be set independently from a pressure value for pellet acceleration. Besides a constant Nd magnet is used for the puncher fixation and no valves are applied to remove propellant gas from the ice generator and barrel. The pellet injectors designs as well as test results are presented and discussed.
Modeling and simulation of time-dependent gas pumping scenarios in ITER

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Recently, an integrated software algorithm for modeling gas distribution systems operating under vacuum conditions has been developed [1]. It has been successfully applied to model the 2012 ITER divertor pumping system and results have been provided for the flow patterns in the cassettes and the divertor ring, as well as for the throughputs in the burn and dwell phases. In all cases the input pressures at the boundaries of the network, representing the pressure distribution in the dome, have been taken to be constant, assuming steady-state conditions. In several ITER gas pumping scenarios however, the data are varying with time and the whole set-up requires a transient approach. A typical example is the dwell phase operation with the dome pressure reducing with time down to a start-up value before the next plasma shot. In the present work, the time-dependent behavior of gas distribution systems subject to transient boundary conditions is simulated. Since the characteristic time in the torus is several orders of magnitude larger than that in the pumping system, the evolution of the flow throughput is modeled in a hybrid manner [2]. At each time step, based on kinetic modeling, a steady-state flow configuration is solved to estimate the amount of gas passing through the network and then the pressure of the vessel is updated by applying the mass conservation principle and the equation of state. As the dome pressure is reduced with time, the flow becomes more rarefied and consequently, the gas flow towards the cryopumps is also reduced, affecting significantly the pump performance and the overall time needed to evacuate the vessel. The elapse times between pulses are computed assuming various initial and base pressures for several pumping scenarios. [1] N. Vasileiadis et al, FED, 103, 125-135, 2016. [2] M. Vargas et al, JVSTA, 32, 021602, 2014.
An innovative approach for DEMO core fuelling by inboard injection of high-speed pellets

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Core fuelling of DEMO fusion reactor is under investigation within the EUROfusion Work Package “Tritium, Fuelling and Vacuum”. An extensive analysis of fuelling requirements and technologies, suggests that pellet injection still represents, to date, the most realistic option. Modelling of both pellet penetration and fuel deposition profiles for different injection locations, assuming a specific plasma reference scenario and the ITER reference pellet mass \(6 \times 10^{21}\) atoms, indicates that: 1. Low Field Side (LFS) injection is inadequate, even at speeds \(\geq 10\) km/s; 2. Vertical injection may be effective only provided that pellets are injected at \(\approx 10\) km/s from a radial position \(\leq 8\) m, so this injection scheme is not presently considered as a practical option, unless such high injection speeds will become available; 3. Effective core fuelling can be achieved launching pellets from the High Field Side (HFS) at \(\approx 1\) km/s. Guiding tracks with a bend radius \(\geq 6\) m are envisaged to deliver intact pellets at 1 km/s. HFS injection was therefore selected as the reference scheme, though scenarios featuring less steep density and temperature gradients at the plasma edge could induce to reconsider vertical injection at speeds in the range of 4 to 5 km/s. The results of above simulations rely, of course, on the hypothesis that pellets are delivered at the plasma edge with the desired mass and speed. However, mass erosion and fracturing of pellets inside the track, severely limiting the transfer speed, as well as pressure build up and speed losses at relevant injection rates, might hamper the use of curved guide tubes. An additional innovative approach, aimed at individuating inboard straight “free flight” injection paths, to inject pellets from the HFS at significantly higher speeds, is proposed and discussed as a backup solution. Outboard high-speed injection is still being considered, instead, for JT-60SA.
P4.162

Dynamic model of ITER cryo-pumps cryogenic distribution system: torus pumping and regeneration scenarios

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The ITER vacuum system, one of the largest and most complex vacuum systems ever to be built, will use first of a kind cryopumps to provide high vacuum conditions to the torus vessel, cryostat vessel, and neutral beam injectors. In order to evacuate the high gas flows required by the plasma scenarios, the cryopumps will need sequential regenerations with unprecedented high frequencies. The Front-End Cryogenic Distribution System (FECDS), made of 12 cold valve boxes (CVB) and 1 Warm Regeneration Box (WRB), will provide helium to each of the cryopumps at the required pressure, temperature and mass flowrate, in order to control the pumps behaviour during operation and regeneration. Such CVBs and WRB will operate in parallel, with dynamic synchronization to satisfy the cryopumps requirements. A model of the FECDS and cryopumps system is being developed within EcosimPro/Cryolib simulation environment, with the purpose of simulating the cryogenic process in dynamic scenarios and optimize the final design and the control of the system. This paper presents the dynamic model of the torus subsystem, made of 6 torus CVBs and the connected cryopumps, during operation and 100K regeneration scenarios, which will imply the staggered dynamic transition of the system components between several states during plasma pulses. The preliminary simulation results will be presented, giving useful information for a refined setting of parameters like pressures, mass flow rates, and valves control. These results are suitable to be used to optimize the FECDS final design and operation.
Development of Cryoadsorption Cryopump & its Related Auxiliary Technologies in India

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Indigenous cryoadsorption cryopump with large pumping speeds gases like hydrogen and helium is developed and a set of experiments performed at the Institute for Plasma Research (IPR), India. Towards its successful realization, technological bottlenecks were identified, studied and resolved. Hydroformed cryopanels were developed from concept leading to the design and product realization with successful technology transfer to the industry. It led to the expertise for developing hydroformed panels for any desired shape, geometry and welding pattern. Activated sorbents were developed, characterized using an experimental set up which measures adsorption isotherms down to 4K for hydrogen and helium. Special techniques were evolved for coating sorbents on hydroformed cryopanels with suitable cryo-adhesives. Various arrangements of cryopanels at 4K surrounded by 80 K shields and baffles (which are also hydroformed) were studied and optimized by transmission probability analysis using Monte Carlo techniques. CFD analysis was used to study the temperature distribution and flow analysis during the cryogen flow through the panels. Integration of the developed technologies to arrive at the final product was a challenging task and this was meticulously planned and executed. Result was a cryoadsorption cryopump offering pumping speeds as high as 50,000 to 70,000 l/s for helium and 1,50,000 l/s for hydrogen with a 3.2m² of sorbent panel area. From R&D to product development has led to establishment of a dedicated lab with design and characterization facilities under one roof. The first laboratory scale pump integrating the developed technologies was a Small Scale CryoPump (SSCP-01) with a pumping speed of 2,000 l/s for helium. Subsequently, Single Panel CryoPump (SPCP-01) with pumping speed 10,000 l/s for helium and a Multiple Panel CryoPump (MPCP-08) with a pumping speed of 70,000 l/s for helium and 1,50,000 l/s for hydrogen respectively were developed. This paper describes realization of journey towards development of product.
P4.164

Mercury ring pump proof-of-principle testing in the THESEUS facility

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The reduction of tritium inventories is a key challenge for DEMO and future fusion power plants. As large amounts of tritium have to be processed in the inner fuel cycle, an inventory-optimized vacuum pumping process – the KALPUREX process – has been developed at KIT. Here, continuously working and non-cryogenic vacuum pump trains will be used in order to keep the tritium residence times and thus the inventories in the pumps small. These pump trains comprise a combination of diffusion pumps and liquid ring pumps, both using the same working fluid mercury. Liquid ring pumps are typically used with a non-tritium compatible working fluid (e.g. water or oil) and at relatively high inlet pressures (some kPa, defined by the vapour pressure of the working fluid). As roughing pumps in fusion, they have to be made fully tritium compatible and optimized in inlet pressure. Both can be done when using mercury as working fluid, as it is perfectly tritium compatible and has a low vapour pressure. As no mercury ring pump was existing on the market and no performance predictions could be made for such a high density working fluid, a commercially available pump has been modified and used for proof-of-principle experiments in the THESEUS facility. In this paper, the design of the KIT mercury ring pump will be described and performance curves, like pump down- und pumping speed curves for different gases, will be presented. Furthermore, operational limits (e.g. ultimate pressure, thermal limitations) will be discussed and suggestions for the design of future mercury ring pumps will be made. Also methods to avoid the migration of mercury vapour outside the pumping system will be shown and validated by corresponding measurements.
Amperometric hydrogen sensors for molten metals

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Accurate and reliable tritium management is of basic importance for the correct operation conditions of the blanket tritium cycle. The determination of the hydrogen isotopes concentration in liquid metal is of high interest for the blanket correct design and operation. Sensors based on solid state electrolytes can be used to that purpose. It is worth mentioning that these type of sensors offer quick and easy to measure signals, high chemical stability and its ionic conductivity increases with the temperature. Potentiometric hydrogen sensors based on solid state electrolytes for molten lithium-lead eutectic were previously studied at the Electrochemical Methods Laboratory at Institut Químic de Sarrià (IQS) at Barcelona. Due to the satisfactory obtained results, amperometric configuration was also evaluated. The probes are based on solid state electrolytes and are considered Proton Exchange Membranes – PEM. These electrolytes are perovskite type materials, where the electrical carriers are positive holes, excess electrons, oxide ion vacancies and interstitial protons which interact with oxide ions. In the present work, the most promising solid state electrolytes for potentiometric sensors have been synthesized in order to be tested as PEM in the amperometric H-probe. Amperometric measurements of the ceramic elements have been performed at different hydrogen concentrations (from 0 mbar to 30 mbar), different temperatures (from 500 °C to 650 °C) and applying different polarization potentials to the sensor.
Development of advanced hydrogen permeation sensors to measure Q2 concentration in lead-lithium eutectic alloy

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A crucial issue for the design of HCLL (Helium Cooled Lead Lithium) Test Blanket Module of ITER and HCLL, WCCL, DCLL Breeder Blanket of DEMO is to efficiently characterise the tritium inventory inside the blanket and the permeation of tritium into the coolant in order to reduce as much as possible the radiological hazard towards the external environment. A fast and reliable sensor is required both for tritium management at the several steps of the reactor fuel cycle and for the development of tritium processing systems, e.g. the TES (Tritium Extraction System). Under these points of view, three advanced hydrogen permeation sensors have been developed on the basis of previous sensors manufactured and assessed at ENEA Brasimone. Two permeation sensors were manufactured, a helical and a cylindrical one made of pure iron, in order to reach acceptable response times. Moreover, a sensor made of niobium with a pure iron capsule coated in palladium was manufactured in order to measure the Q2 concentration in PbLi in Dynamic mode with high reliability. The sensors performance is assessed in liquid phase thanks to the Hydrogen Permeation Quartz Chamber (HYPER-QUARCH) device installed at ENEA C.R. Brasimone. Using hydrogen instead of tritium, several tests are carried out in both dynamic and equilibrium mode, simulating ITER and DEMO operative conditions. The tritium concentrations measured are evaluated using different values of the Sievert’s constant.
Improvement of quantitative analysis method of tritium using hydrophobic catalyst

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Various methods of tritium measurement have been applied depending on a chemical form of tritium. A method combined oxidation catalyst and water bubblers has been used as one of the most quantitative analysis methods for gaseous tritium. We previously developed a quantitative analysis system to measure gaseous tritium in a high accuracy using an organic-based hydrophobic platinum catalyst. However, the previous tritium analysis system has a shortcoming that the organic-based catalyst has no resistance to a high temperature of more than 100°C, so that the system was not able to measure combustible gas such as tritiated hydrocarbons because the heat of reaction of hydrocarbons combustion on catalyst was large. Then, we developed a hydrophobic platinum catalyst having heat resisting property and high activity to measure various chemical forms of tritium. A new quantitative analysis system using the hydrophobic catalyst enhanced the thermal durability performed as good as the previous system over wide temperature range. The developed tritium measurement system is applicable widely for various tritium measurements, estimation of tritium generation rate at TBM, tritium behavior in various materials and so on. Details will be reported at the conference and the paper.
Tritium introduction module design for the JET tokamak


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In support of ITER, two experimental campaigns are foreseen to take place at JET, the first with tritium only and a second with deuterium plus tritium to explore the machine fusion potential. To support the tritium operation, a total of five Tritium Introduction Modules (TIMs) are expected to be installed at JET, one on top of the machine, another in the mid-plane and three in the divertor region. Since no human intervention or serviceability to the TIMs is foreseen during the tritium experimental campaigns, their design needs to incorporate redundancy, reliability, secure operation and conform to JET specific design, inspection, testing and safety case requirements. The challenges to the design in complying with operational requirements, connection to the Active Gas Handling System (AGHS), predicted neutron and gamma levels, were taken into account and shaped the design choices presented here. The function of a TIM is a controlled tritium injection to the vacuum vessel while preventing accidental release to vessel or environment. TIMs consist of a secondary containment vessel, internally filled with a purge gas of recirculating Nitrogen, up to 3 bar. This vessel also encloses the primary tritium containing components, including the piezoelectric valves, the primary tritium reservoirs (each holding 1g at 800 mbar abs), pressure and temperature instrumentation for accurate tritium inventory accounting. Prototype piezo-valves developed by VAT Vacuumvalves AG with a stroke of 160μm and orifice area of 5.2mm², are tested for a flow rate of 1.6 Bar.L/s. The TIMs are connected through a supply network involving distribution vessels and transfer lines connecting to the AGHS valve box and ultimately to the primary Uranium bed storage and to the monitored discharge stack for the nitrogen exhaust. This paper describes the module design and details the operational and safety case requirements considered for safe Tritium operation.
Operational aspects of the JET Tritium Introduction Modules

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As part of the JET Programme in Support for ITER, campaigns with pure Tritium-Tritium (TT) fuel and Deuterium-Tritium (DT) mixture are planned at JET. Unlike the previous DT campaign at JET, these campaigns require a much higher tritium flow rate, particularly, the TT campaign can require up to 3.7 grams of tritium on a single pulse. Five tritium introduction modules (TIMs) fed from the Active Gas Handling System (AGHS) are to be installed; three on the divertor region, one in the mid-plane and another on outer top of the machine. The TIMs location, design, operational characteristics were chosen so that most Deuterium experiments can be matched in Tritium. Since the access to the machine will be severely limited, the TIMs were designed to accommodate a reliable and redundant operation while maximising the operational space. Each TIM includes two five litre reservoirs and two high flow piezoelectric valves (1.6 Bar.L/s) that can be operated together or independently, thus maximising the system flexibility and compatibility with different experiments requirements. The low reservoir volume combined with a high flow valve requires a precise control of the valve opening in order to reproduce a pre-set gas flow waveform, this is achieved by adjusting the opening request according to the reservoir depletion throughout the pulse. Several use cases are described in the paper.

In addition Tritium accountancy will play a critical role in the future Tritium campaigns. In total 60 grams of Tritium will be used and recycled through the AGHS. Safety considerations limit the quantity of Tritium inside the JET tokamak to 15 grams, implying that the tritium calculation pre-pulse, correct operation during the pulse and accountancy after it have to be accurate and reliable. This paper provides a guidance on the Tritium inventory monitoring system while detailing the operation sequence.
Enhanced Jet Stability For The Melt-Based Production Of Lithium Orthosilicate/Metatitanate Pebbles

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Advanced tritium breeder pebbles, composed of lithium orthosilicate with additions of lithium metatitanate as a secondary strengthening phase, are produced using a melt-based process. Synthesis powders are heated to high temperatures in a platinum alloy crucible, forming a melt, which is then ejected through a nozzle to form a laminar jet. Longitudinal surface instabilities cause the disintegration of the jet into droplets as described by the Plateau-Rayleigh instability theory. The droplets are subsequently solidified using liquid nitrogen to form the pebbles. A high-speed camera is used to record the droplet formation dynamics at 3500 FPS (frames per second) and offline analysis is performed to determine various jet characteristics including the droplet generation frequency and the jet velocity, from which the instability wavelength can then be derived. Due to various problems when determining rheological properties of melts at high temperatures, optimum process parameters are usually determined empirically. However, according to Rayleigh, the wavelength on the surface of the jet can also be used to characterise the jet stability and determine the optimum droplet formation parameters. As the operating pressure is the only adjustable parameter during the production ‘jetting stage’ which affects the jet dynamics, it was varied during the production of pebbles at standard temperatures and subsequently the relationship between the operating pressure and wavelength was determined. Additionally, samples were manually extracted at each operating pressure to test the practicality of the study, as well as the relevance of the optimum wavelength as described by Rayleigh’s theory. The results were used to optimise the jet dynamics and subsequently to improve the yield and the pebble size distribution by establishing an optimum operating pressure.
The test cell configuration under IFMIF-DONES condition

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As the complementary work of IFMIF-EVEDA (International Fusion Material Irradiation Facility-Engineering Validation and Engineering Design Activities) project, WPENS (Work Package Early Neutron Source) project in the framework of EUROfusion activities is committed to the engineering design of an IFMIF-DONES (Demo Oriented Neutron Source) facility, which is an accelerator based intense fusion-like neutron source with the objective of qualifying structural materials that will be used in the DEMO fusion reactor. The test cell (TC) in IFMIF-DONES is the central place as the meeting point of the three major systems (test systems, lithium systems, and accelerator systems) to host the irradiation test module, lithium target assembly (TA) and the end section of the accelerator. Although the reference TC design of IFMIF-DONES, in a large extent, inherits that of IFMIF-EVEDA, design justifications have been proposed with the consideration of the issues on stability of the lithium flow, activation and maintainability of key components, configuration of TC biological shielding, arrangement of in-cell components, impacts on other systems, dose rates in adjacent rooms, and etc. Special attentions are put on the location selection of the lithium quench tank (QT), the shape of the connection channel between the TA and the QT, and the configuration of TC floor. In this paper, the advantages and drawbacks of the reference TC design as well as the proposed design justifications have been investigated and discussed based on updated CFD and neutronic calculations under IFMIF-DONES condition. As the output of the investigations and analysis, a converged IFMIF-DONES TC design has been derived and is described in this paper.
Measurement of Free-surface Lithium Flow using Laser Reflection Method

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In the international fusion materials irradiation facility (IFMIF), 14 MeV neutrons are generated by 40 MeV deuterion beam injection into a high-speed liquid lithium (Li) plane jet, flowing along a vertical concave wall in vacuum. Measurement of a free surface flow and fluctuation of the thickness are required to produce a stable neutron field and maintain the safety of Li target system. In previous study, we proposed laser reflection method as new non-contact measurement of the Li surface fluctuation. The method is a technique to measure fluctuation of jet thickness from a reflection point of laser beam which represents a slope angle of fluid surface. The method can get the time variation of Li surface fluctuation in contrast to the contact probe method. The present work is intended to provide new insight into characteristics of Li surface wave from the time variation of surface fluctuation which obtained from the method. Experiment is performed using the Li circulation loop at Osaka University. Wave characteristics are investigated from the surface fluctuation by the crest-to-trough method. The wave height and a half wave period is defined as the height and time interval between crest and trough, respectively. First, probability distribution of wave period of each Li flow velocity is calculated. As the result, it is found that short period wave increase with increasing velocity. In addition, the distribution has log-normality. It is similar to the characteristics of the previous study using the contact probe. Next, it is known that waves break when their steepness (height per wavelength) exceeds the critical steepness. The critical steepness of regular water wave is 0.14107. As the result of investigation of the relation between wave height and wavelength of this experiment, the critical steepness of Li wave is smaller than that of water wave.
Numerical investigation of cavitation phenomena in the free surface liquid-lithium flow

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The configuration of the Early Neutron Source (ENS) is the IFMIF-DONES (DEMO Oriented Neutron Source) approach, based on an IFMIF-type neutron source. It aims providing an intense fusion-like neutron spectrum with the objective to qualify on an accelerated time scale structural materials to be used in the future DEMO fusion reactor. IFMIF-DONES is based on the interaction of single 40MeV 125mA deuteron beam impacting a flowing liquid lithium target to simulate DEMO like neutron flux spectrum for fusion material irradiation experiments. The lithium free-surface flow providing the desired neutron flux and heat removal is operated in vacuum. Hence its operational performance determines the design of the target assembly and interfaces such as e.g. the quench tank. Since it is practically unfeasible to avoid geometrical discontinuities such as steps, obstacles or gaps in engineering designs, a realistic assessment in terms of magnitude and location of the potential cavitation risk in the lithium system components is needed. The present work focuses on the numerical investigation of cavitation phenomena in the lithium flow at IFMIF-DONES relevant operation conditions. Calculations reproduce different cavitation processes depending on the kind of the wall surface discontinuity. In case of the flow over the lateral gaps, in the channel lithium gaseous phase generated within the gap remains stable and does not collapse. Simulations of the lithium flow over the backward-step show the generation of the gaseous lithium phase within the flow separation area and formation of a stable sheet cavity on the wall surface. The subsequent breakup of the sheet cavity in the flow reattachment region is accompanied by generation and collapse of unstable vapor structures downstream. The risk of cavitation induced erosion on the wall surface is assessed using a function based on the mean value of the time derivative of the local pressure.
P4.175

Cavitation upstream of liquid lithium target for intense fusion neutron source

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A liquid-Li free-surface stream flowing at 15 m/s under a high vacuum of 1E⁻³ Pa is to serve as a beam target (Li target) for the planned International Fusion Materials Irradiation Facility (IFMIF). The Engineering Validation and Engineering Design Activities (EVEDA) for the IFMIF are implemented under the Broader Approach. As a major activity of the Li target facility, the EVEDA Li test loop (ELTL) was constructed by the Japan Atomic Energy Agency. In a validation experiment using the ELTL, the stable Li target under IFMIF conditions (Li temperature: 250 deg.-C, velocity: 15 m/s, vacuum pressure: 1E⁻³ Pa) has been demonstrated so far. This study focuses on cavitation-like acoustic noise which was detected in a conduit upstream of the Li target. This noise was detected by using acoustic-emission sensors (AE-900S-WB, NF Corp.) that were installed at several locations upstream and downstream of the target assembly via acoustic wave guides (stainless steel rods). The intensity of the acoustic noise was measured versus flow rate of the Li target in several gaseous pressure cases. A time-frequency analysis by Continuous Wavelet Transform (CWT) for the acoustic signal was performed to characterize the acoustic noise, which determined the cause of the acoustic signal was cavitation. In addition, the occurrence of cavitation at the highest point of the pipe upstream of the target assembly was theoretically discussed by using Bernoulli’s equation and compared with the experimental observations. As a result, we revealed a proper start-up pressure of the Li target to avoid cavitation upstream of the target assembly, which will be a design basis for the future IFMIF.
Numerical study on detailed flow structure inner high-speed liquid metal lithium jet

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A high-speed liquid metal lithium jet (Li jet) with a free surface is planned as a target irradiated by two deuterons beams to generate a neutron field in an accelerator based neutron source, such as that in the international fusion materials irradiation facility (IFMIF). In the IFMIF, it is desirable to stabilize the Li jet for the efficiency of the neutron generation and the safety of facilities, and we have been thus studying characteristics of the Li jet experimentally and numerically. Especially, in order to understand observed phenomena experimentally, it is important to obtain the information on the inner flow structure of the opaque Li jet using a simulation. The Li jet flows out from a two-staged contraction nozzle, which is used for reducing a thickness of a boundary layer of the Li jet at the nozzle exit. In previous numerical study, it was found that longitudinal vortices due to hydrodynamic instability at concave walls were generated inside the boundary layer of the nozzle in the Li jet velocity of 15 m/s. In addition, the random pattern of the surface wave on the Li jet in this case was also caused by both the relaxation of the free shear stress layer under the free surface and these vortices mentioned above. In this paper, we will present results of the simulation model extended to 100 mm downstream from the nozzle exit in the jet part length and 5 mm in the model width. As a result, it was found that vortices under the Li jet surface and at the bottom of the Li jet were coalesced and dissipated as flowing downstream and that the random surface fluctuation could be confirmed at points of 55 and 95 mm downstream from the nozzle exit as well as experimental results.
Study on suppression of surface fluctuation of liquid Li jet by magnetic field

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Liquid metal flow has been expected to be applied in various fields. For example, sodium and lithium (Li) are applied as a coolant in the fast-breeder reactor and space nuclear reactor, Li jet as a beam target in the International Fusion Materials Facility (IFMIF) and as a charge stripper in Radioactive Isotope Beam Facility (RIBF) at RIKEN, lithium-lead (Li-Pb) as a liquid metal blanket in a Helical-type fusion reactor and so on. Especially, since the Li jet must flow with high velocity for a heat removal in the IFMIF and the RIBF, the surface of it might fluctuate. The surface fluctuation causes the increasing of the damage risk of the flow channel and the moderation of the generated neutron in the IFMIF, and the diffusion of beam profile passing through the fluctuating Li jet leads to the low beam intensity in the RIBF. So, in these facilities, it is strongly desirable to suppress such surface fluctuation of the Li jet. In this study, we verify suppression of the surface fluctuation of it by Magneto-Hydro-Dynamics (MHD) effect. Results in this research on MHD effect to liquid metal contribute to the development not only of a beam target in the IFMIF and RIBF but also of the Li-Pb blanket, in which the heat transfer under strong magnetic field could determine the performance. In this study, we designed and made magnet-insertion apparatus for applying a magnetic field to the Li jet in the liquid Li circulation loop at Osaka University, and the experiment with this apparatus was conducted. The insertion system consists of a magnet part for applying magnetic field with two Samarium-Cobalt magnets, an iron core and the up-and-down driving system. In this experiment, by driving the magnet part, the surface shape with/without the magnetic field could be observed.
Mechanical testing of the IFMIF HFTM-DC prototype during operation in the HELOKA-LP helium loop

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The so called High Flux Test Module (HFTM) represents the component of IFMIF (International Fusion Irradiation Facility) in which material specimens are being placed that accumulate the highest neutron induced damage rates ($\geq 20$ dpa/fpy). Damage rates of this magnitude are limited to a volume of $\sim 500$ cm$^3$ (attenuation in beam direction) behind a beam footprint of 20x5 cm. The high flux region of the module is contained in a flat faced, cuboid volume of 5.6 cm depth. Efforts for a high spatial utilization and the demand for a high neutron transmission require a thin-walled container design. An array of mini-channels (1 mm gap) through which low pressure (0.3 MPa) helium gas flows was chosen as an efficient, space-saving method to cool the container and the material specimens. Due to place constraints between the target and the other irradiation modules the HFTM is implemented as a slender and long construction with features that are challenging for pressurized equipment. Experimental studies on a 1:1 prototype of the HFTM-DC (double compartment) were conducted in the Helium Loop Karlsruhe – Low Pressure (HELOKA-LP) during 2015. The experiments also included intensive testing to demonstrate the mechanical reliability of the HFTM under IFMIF relevant operation conditions. Therefore, the module is instrumented with numerous sensors which measure displacement, deformation and mechanical strain. The reactions on temperature and pressure loads were studied. In this paper the experimental results will be presented and compared to the numerical (FEM) simulation studies.
Steady-state and transient thermal-hydraulic performance of the IFMIF High Flux Test Module

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During the EVEDA phase of the International Fusion Materials Irradiation Facility (IFMIF), the High Flux Test Module (HFTM) was developed as dedicated irradiation device for Small Specimen Test Technique. In the intensive IFMIF neutron radiation field the specimens are contained in temperature controlled irradiation rigs. Since one of the requirements for the HFTM is to provide a uniform temperature field for the irradiated specimens, thermal testing was a priority for the performed validation activities. In the HFTM “single-rig” (HFTM-SR) experiments a single rig of 1:1 scale was tested, while the neighboring rigs were modeled with heater plates. The heater plates and the specimen region inside the rig were instrumented with thermocouples to monitor heat transfer and specimen temperature spread. In the High Flux Test Module “double compartment” (HFTM-DC) experiments a fully equipped prototype with three heated rigs was tested in the HELOKA-LP helium loop. Heater cartridges are used to substitute the nuclear heating. These experiments show that the full range of operation temperatures (250 - 550°C) required for the IFMIF HFTM could be well achieved and well controlled with and without “nuclear” heater power. The temperature spread measured inside a capsule is in the range of +/-3K in the lateral direction and +2K / -8K in the vertical direction for the 350°C reference case. This compares to an allowed +/-19K according to the requirements. No unforeseen thermal hydraulic effects like oscillations, hysteresis etc. could be detected. To cool down from 350°C to 50°C it takes roughly 315 seconds, heating up 135 seconds. Additional the temperature and strain on the container was measured for the tested temperature levels.
Measurement of transient flow characteristics of target flow in water experiment for IFMIF

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International Fusion Material Irradiation Facility (IFMIF) is the facility generating the high flux and high energy neutron to develop a material for a nuclear fusion reactor. In the IFMIF, high-speed liquid lithium (Li) jet is used as the target irradiated by two deuteron beams. Since the Li jet must flow with high velocity for the heat removal, it is important to research on the characteristics of Li flow. Laser probe method has been developed as a promising measurement method of target flow characteristics with non-contact measurement and demonstrated as one of the task in IFMIF/EVEDA. These researches have been aimed toward the steady-state flow characteristics of the Li jet. On the other hand, in the actual IFMIF, it is also necessary to clarify transient flow characteristics at start and stop of the system for the operation because of the target flowing along with a vertical and a concave flow channel. In this study, water experiment to obtain them at start and stop is thus conducted using laser probe method. Water can be substituted for liquid Li as the target, because the kinematic viscosity of the Li at the operation temperature in the IFMIF is nearly equal to that of water at normal temperature and pressure. Water loop used in this experiment mainly consists of a two-staged contraction nozzle, a vertical and a concave flow channel, a gas release valve, a pump and a buffer tank. In addition to laser probe, high-speed video camera is used to observe the flow pattern of the water jet. As a result, it was confirmed that the surface fluctuation at start and stop of flow became larger than that at steady-state. The flow at the stop also became stable by venting gas from the release valve.
Neutronics assessment of different quench tank location options in IFMIF-DONES

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The location of the lithium quench tank (QT) is an important safety related issue in the design of the test cell (TC) of the IFMIF-DONES (International Fusion Material Irradiation Facility-DEMO Oriented Neutron Source). In the current reference design, the QT is situated outside the TC and is connected to the target assembly through a long lithium outlet channel penetrating the TC floor. Compared to the option with the QT located inside the TC, this configuration significantly reduces the activation of the QT and the tritium production. However, the long lithium channel may cause instabilities of the lithium flow and will result in a strong neutron streaming through the channel into the lithium systems room beneath the TC floor. Another recent proposal considered to locate only the upper half of the QT inside the TC while the lower half is embedded in the TC floor which is then reinforced with additional shielding. Such a configuration would have both advantages and disadvantages which need to be evaluated on the basis of a dedicated analysis. In this work, neutronics analyses are carried out to assess both options of the TC locations in IFMIF-DONES. The McDeLicious-11 code, which is an extension of the MCNP5-1.6 Monte Carlo code with the capability to simulate the deuterium-lithium neutron source in IFMIF-DONES, is employed in the calculations. The neutron flux distribution, the tritium production rate and the dose rate distribution during operation are evaluated and compared. In addition, the shut-down gamma dose rate distribution in the lithium systems room is calculated. This is of great concern for maintenance operations scheduled for the lithium-loop facilities. The shut-down dose rate calculations are carried out by using the R2Smesh code system developed at KIT. Based on the results of these analyses suggestions are made for the QT arrangement in IFMIF-DONES.
Manufacturing of the IFMIF HFTM double compartment prototype

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The High Flux Test Module (HFTM) of the International Fusion Materials Irradiation Facility (IFMIF) is a device to enable irradiation of Small Scale Testing Technique (SSTT) specimens by neutrons up to a structural damage of 50 displacements per atom (dpa) in an irradiation campaign of one year. The IFMIF source generates neutrons with a D-T-fusion-relevant energy spectrum and a flux to achieve a damage rate over 20 dpa per full power year in a theoretical volume of 0.5 l. Irradiation temperatures are required in the range of 250 - 550°C. According to the IFMIF conditions and requirements, the IFMIF HFTM has been developed in the IFMIF/EVEDA phase and a prototype was constructed and tested. The manufacturing process of relevant parts, like attachment adapter and container, is presented – especially in regard of problems in manufacturing accuracy. The capsule manufacturing process in regard of brazing process and finishing of the capsule shape is explained in detail. Further the instrumentation of the prototype for experimental data generation is presented. Optimization potentials derived from the manufacturing process and the experimental experiences are highlighted.
The Conventional Facilities of the Linear IFMIF Prototype Accelerator (LIPAc)

PRUNERI, Giuseppe

The Conventional Facilities of the Linear IFMIF Prototype Accelerator (LIPAc) Authors G.Pruneri, P.Cara, R.Heidinger, A. Kasugai, J. Knaster, S. Ohira, Y.Okumura, K.Sakamoto, and the LIPAc Integrated Project Team. The International Fusion Material Irradiation Facility (IFMIF) aims at qualifying and characterising materials capable to withstand the intense neutron flux originated in the D-T reactions of future fusion reactors thanks to a neutron flux with a broad peak at 14 MeV capable to provide >20 dpa/fpy on small specimens also qualified in this Engineering Validation Engineering Design Activity (EVEDA) phase. All its broad mandate has been successfully achieved, the only pending, is the validation of its Accelerator with its conventional Facilities. The validation of IFMIF’s accelerators will be achieved in this on-going phase until December 2019 with the operation of a deuteron accelerator at 125 mA CW mode and 9 MeV, which is presently under installation and commissioning in Rokkasho (Japan). The target availability of IFMIF facility, 70%, is one of its main challenges since demands extraordinary individual availabilities of the sub-systems, like the accelerator with 87%. LIPAc, the Linear IFMIF Prototype Accelerator presents a broad spectrum of ancillary equipment to optimize its operational beam time. A description of the Nuclear HVAC of IFMIF has already been reported [1]. The present paper describes the design of LIPAc of the Conventional systems among which we address the Electrical Power Supply, the Heating Ventilation & Air Conditioning (HVAC), the Heat Rejection System (HRS), the Service Water System (SWS), the Service Gas System (SGS), the Cryoplant System (Cryo) and the Fire Protection System (FPS). [1] G. Pruneri et al., Design principles of a nuclear and industrial HVAC of IFMIF, Fusion Engineering and Design 103 (2016) 81–84
In the conceptual design of the beam dump shielding for the foreseen fusion-relevant irradiation facility IFMIF, an inner lead cylinder performs the shielding of the highly activated copper cone undergoing the deuteron beam bombardment and low-alloy steel is used for front shielding. In order to reduce the residual dose around the beam dump at beam-off conditions and dose at hands-on operations, lead as an alternative material for the front shielding is considered. The photon dose originated in the activation of the main components of such materials and of their impurities is calculated with the use of SEACAB system following the rigorous-two-step method (R2S) including the use of MCNPX code for neutron and photon transport and ACAB code for neutron activation. This methodology has been developed internally and benchmarked with the most accepted fusion-relevant benchmark for residual dose performed at Frascati Neutron Generator facility. The source of secondary neutrons produced by the interaction of deuterons on the beam stopper has been calculated by the UNED team with the use of MCUNED and TALYS 1.0. The MCNPX input has been prepared starting from CAD files supplied by CIEMAT and translated to MCNP format with the use of MCAM interface developed by FDS team.
Numerical study on fluid dynamics of liquid metal breeder under magnetic field

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It is desirable to develop liquid lithium-lead (Li-Pb) blanket for helical-type fusion reactor because of its high cooling and tritium-recovering abilities. Since heat transport under a strong magnetic field in a fusion reactor determines the performance of liquid metal blanket (LMB), it is important to clarify the mechanism of the interaction between Li-Pb flow and the magnetic field. On the other hand, we have been studying flow characteristics of liquid Li jet using the liquid metal Li circulation facility at Osaka University experimentally and using computational fluid dynamics (CFD) code, ANSYS FLUENT numerically for IFMIF/EVEDA. Therefore, in this study, we aims to contribute to the development of LMB by clarifying magnetohydrodynamic (MHD) characteristics of liquid metal flow using the knowledge obtained from studies for IFMIF/EVEDA. In previous study, the simulation model of Li flow inside the nozzle of our Li loop was validated about the flow distribution by comparing with the water experiment, and thus the performance of MHD model equipped with ANSYS FLUENT was confirmed by the validated nozzle flow model as a starting point in this study. The turbulence model was Large Eddy Simulation (LES). It was assumed that uniform external magnetic flux was applied to the direction perpendicular to flow direction overall and the magnitude was set to 0.3 [T]. As a result, it was confirmed that vortices generated inside a boundary layer near a wall were clearly suppressed under the magnetic field and eddy currents were also generated near the wall. Then, we simulate Li-Pb flow with rectangular channel, which has an insulating wall, under non-uniform magnetic field based on this model. In this simulation, Reynolds number and Haltmann number are changed as a parameter to determine simulation conditions. This paper will also present the results of this simulation.
Modelling tritium permeation during PbLi capsule irradiation

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Envisioned fusion facilities for energy production are currently under development within EUROfusion program. In these devices, a D-T plasma is used as energy source. While deuterium is abundant, tritium has to be produced on-site. Tritium, as one of the hydrogen isotopes, easily diffuses through metallic walls of its confinements. Such ‘tritium leakage’ can be limited by developing an efficient anti-permeation barrier. For the purpose, coatings based on Al2O3 are under thorough investigation. To determine their efficiency in decreasing the tritium losses, permeation tests are carried out in a variety of conditions replicating different aspects of the process. One of such tests is an in-pile irradiation of a capsule filled with Pb-17Li alloy and coated sample tubes. The aim of the test is to verify the effect of radiation on tritium permeation rates. To evaluate the tests, a mathematical model describing tritium permeation within the complex geometry of the PbLi capsule is being developed. In this contribution, a parametric study of the tritium permeation within the PbLi capsule is presented.
Monitoring of oxide layer structures in Pb and Pb-Li alloy by electrochemical impedance spectroscopy

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The development of functional layers such as the tritium permeation barrier and the anti-corrosion barrier is one of the important issues for the development of liquid breeder blanket. The functional layers with the self-healing function have been developed based on the mechanism of the oxide layer formation. The oxides of yttria (Y2O3) and zirconia (ZrO2) have an excellent chemical stability. The ceramic coating by these oxides showed the good performance as the permeation barrier for the hydrogen isotopes at high temperature. These layers can be fabricated by the pre-oxidation treatment. Therefore, these layers fabricated by the oxidation of the metals are the candidate of the self-healing type functional materials. However, the chemical stability in the liquid Pb and Pb-Li alloy are not made clear. In the same time, the methodology of the online monitoring for the time-related deterioration of these functional layers in the harsh environment must be established. It was found that the thickness and the structure of the oxide layers could be monitored using electrochemical impedance spectroscopy (EIS) in the previous study. The purpose of the present study is to investigate the chemical stability of Y2O3 and ZrO2 in liquid metal Pb and Pb-17Li. The corrosion of these layers in the liquid metals was monitored on-line by EIS method. Pre-oxidation treatment for the rod type specimens of yttrium (Y) and zirconium (Zr) were performed at 773K for 350 hours in air environment. Then, the specimens are immersed in the liquid metals at the temperature between 773K and 873K. The impedance response for the preformed oxide layer was obtained by the EIS measurement.
Experimental loop and purification system design for corrosion in flowing PbLi

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The use of PbLi and RAFM steels in blanket applications requires a better understanding of material compatibility related to physical/chemical corrosion phenomena in the 450-550°C temperature range. The impact of corrosion includes deterioration of the mechanical integrity of the blanket structure due to the wall thinning. Furthermore, serious concerns are associated with the transport of corrosion products by the Pb-Li coolant constituting strong limitations for the blanket design where the requested high temperatures must be compatible with acceptable corrosion behaviour. Available experimental data on the mass loss for ferritic/martensitic steels in the flowing PbLi spread over a wide range, predicting possible wall thinning (at temperatures higher than 450 °C) from 5 mm/yr up to a few hundred mm/yr. In order to improve the knowledge of corrosion behaviour, further experiments are needed. In this frame, the present work aims to illustrate the design of a new experimental facility named LIFUS II (LIthium for FUSion II) intended to extensively investigate corrosion mechanisms related on coated (Al2O3 based) and uncoated EUROFER samples at constant temperature of 550°C, for three different velocities (0.01, 0.1, 1 m/s) and four different exposure times (1000, 2000, 4000, 8000 h). Furthermore, a “cold trap” purification system is designed in order to remove impurities and corrosion product (resulting from the corrosion of the samples and the steel structures themselves) dissolved in the liquid metal via upper concentration limits imposed by temperature-dependent solubility constrains. The proposed design basically consists in a heat and mass transfer device, where supersaturated solution of impurity is generated as a result of coolant cooling. The performances and efficiency of the purification system will be also assessed in the LIFUS II experimental campaign.
Absorption-Desorption Models for the Determination of Transport Parameters of Hydrogen Isotopes in Eutectic PbLi

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The importance of the hydrogen isotopes transport parameters of Sieverts’ constant and diffusivity in the eutectic lead lithium alloy is well known, as long as it is vital for the determination of tritium management strategies at liquid-metal breeding blanket systems [Helium Cooled Lithium Lead (HCLL), or Dual-Coolant Lead-lithium (DCLL)]. Tritium transport parameters as solubility and diffusivity will determine the magnitude and kinetics of the induced tritium flux in the breeding region from the blanket to the helium cooling loop in the reference breeding blanket systems. These parameters also have to be taken into account in the design of future tritium extraction systems of the breeding alloy or in the He coolant purification system. A theoretical model has been developed to describe the interaction between hydrogen isotopes and the eutectic PbLi alloy and to derive the corresponding experimental values of diffusivity and Sieverts’ constant. This model has been developed for the particular boundary conditions of the Absorption-Desorption facility working at UPV/EHU. Both experimental stages of gas absorption and gas desorption, separated by a thorough pumping-down stage, are entirely simulated in order to reduce the very wide band in the available experimental data (two orders of magnitude) obtained by different research groups using different experimental techniques. The goodness of the theoretical model has been proved by the accurate reproduction of the absorption and desorption experimental signals obtained in the facility with sample temperatures between 523 and 773 K and gas pressures between 1 and $10^5$ Pa.
Lithium conducting ceramics for amperometric sensors in molten metals

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Lithium 6 is the substance 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 prove 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 are stable compounds which can withstand the harsh chemical environment of the melts, the 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 Químic 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, LiYO₂ and Li₄SiO₄ were synthesized in order to be tested as solid state electrolyte for Li-probes. Amperometric measurements of the synthesized ceramic elements were performed at different lithium concentrations using lithium molten salts as lithium containing samples.
P4.193

Corrosion studies of nuclear fusion reactor materials in flowing nanofluid

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In this study, rotating experimental devices were built to investigate the compatibility of the fusion reactor materials RAFM steel, 316L(N) steel, CuCrZr alloy with the Al₂O₃–water nanofluids. Based on the ITER water-cooling program, the experimental condition parameters were fluid velocity of 1.13 and 3.71 m/s, fluid temperature of 70±1°C, testing duration of 2136h, nanofluid mass fraction of 0.01 wt.% and 1 wt.%. The observation and analysis in compositions by SEM, EDS, XPS for the specimen surfaces were performed. The preliminary results indicate that the compatibility of RAFM steel and 316L(N) steel with nanofluids is better than that of CuCrZr alloy. The surface of CuCrZr alloy is covered by oxide film with holes and cracks. Furthermore, the surface morphology of CuCrZr alloy in nanofluids is strongly dependent on the testing duration, flowing velocity and mass fraction. The corrosion mechanism is dominated by oxygen absorption corrosion and erosive-corrosive wear.
ASTEC simulations of dust resuspension in fusion containments compared with the “STARDUST” experimental data

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The ASTEC code is a lumped parameter code originally designed to perform safety analysis in fission nuclear power plants. Recently some modules of ASTEC have been modified by IRSN to be applicable for the safety analysis in the nuclear fusion plants. In particular the CPA module (for the thermal-hydraulics of the containment) and the SOPHAEROS module (to model the physical phenomena involving the aerosols and the vapors and the chemical reactions) have been updated to simulate dust resuspension phenomena in containments, such as the vacuum vessel and the neighbors volumes. It is possible to choose two different models: the “force balance” model that considers a “mechanic” approach, and the rock’ and roll model for which the particles are re-suspended from the surface when they have gained sufficient vibration energy to escape from the adhesive potential well. To test the effectiveness of the models in the peculiar conditions existing in a Tokamak, simulations of dust resuspension have been carried on using the features and the experimental data obtained from experiments performed in the past at the “STARDUST” facility (Small Tank for Aerosols Removal and Dust facility) placed at the University of “Tor Vergata”. The final scope is to test the capability of the code to deal with the dusts resuspension phenomenon in near vacuum conditions. This facility is a small cylindrical vessel in which it is possible to simulate loss of vacuum accidents (LOVAs) at sub-atmospheric pressures pumping air from different valves. To support ASTEC calculations ANSYS Fluent is applied to evaluate properly the flow field and the thermal-hydraulic parameters during the transient. The first results are encouraging but substantial modifications in the resuspension models are necessary to take in account the particular physical conditions: the extremely low density and very high air velocities during the transient.
3D CFD Simulations of dust mobilisation in STARDUST-U facility

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One of the main concerns in Tokamak operation is the dust resuspension and fallout in case of LOVA (Loss Of Vacuum Accident) and LOCA (Loss Of Coolant Accident), as the metallic powders contained in the vessel are radioactive and therefore harmful. Furthermore, they can react explosively with the incoming oxygen if the local composition falls inside the flammability interval and if a hot point triggers the reaction. CFD (Computational Fluid Dynamics) multiphase simulations have been carried out for several pressurisation rates and various inlet points to determine the flow features and the amount of mobilised dust. The numerical simulations results have been compared with experimental data, all the results will be discussed in this work.
Imaging of dust mobilized inside STARDUST-Upgrade facility in case of loss of vacuum accidents

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STARDUST-U facility is an experimental facility voted to help the scientific community to better understand the problem of dust re-suspension and mobilization in case of Loss Of Vacuum Accidents (LOVAs) or Loss Of Coolant Accidents (LOCAs) inside the next generation fusion reactors like the International Thermonuclear Reactor (ITER) or the Demonstration Power Plant (DEMO). In this work the authors will test the capability of STARDUST-U with two experimental set-ups. In the first one, Particle Image Velocimetry (PIV) technique will be used, while in the second one a Shadowgraph technique will be implemented. The results of the experimental campaign will be elaborated with custom software developed in LabVIEW and will be critically analyzed by the authors. These results are fundamental to give the boundary conditions for the numerical simulations in order to develop and validate a multi-phase model to predict dust re-suspension in an enclosed environment.
The influence of dust characteristics on re-suspension: test with tungsten and data discussion

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The future nuclear plants like ITER, DEMO or PROTO are interested by the problems of dust creation and resuspension. Radioactive dust, if resuspended by accidents in the vacuum vessel, can be dangerous because of its toxicity and capacity to explode under certain conditions. The authors have been working since 2006 on dust resuspension problems through the STARDUST facility before and the STARDUST-U facility now. During the experimental campaign they have widely related the thermos-fluidodynamic conditions inside the facility during the accidents reproductions with the dust resuspension. In this work the author have performed scanning electron microscopy (SEM) and X-Ray diffraction (XRD) analyses of the dust (used for the experiments) after different temperature and pressure cycles. The results of these analysis will be compared with the previous data and critically presented by the authors.
Minimising Operator Neutron Dose During JET Shutdown using Virtual Reality

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Effective data visualisation is a key part of the scientific process with complex geometric datasets. It is the bridge between the quantitative content of the data and human intuition. Immersion in virtual reality (VR) provides benefits beyond the traditional “desktop” visualization tools and it leads to a demonstrably better perception of dataspace geometry, more intuitive data understanding, and a better retention of the perceived relationships in the data. VR has a great potential for fusion research and in the analysis of its complex physics-based datasets. The work presented in this paper is one of the applications that have been implemented at UKAEA which combines shutdown dose calculations with a virtual reality model of the Joint European Tokamak (JET). The shutdown dose calculations were performed for various time steps during the JET DTE2 campaign using the UKAEA code MCR2S which links MCNP and FISPACT-II using the rigorous two step method. The 3 Dimensional dose and activation data created by MCR2S is integrated into the game engine Unity using C# routines. A virtual reality model has been created from JET CAD data combined with the 3-dimensional radiation and activation data to create of virtual model of the radiological environment after the DTE2 campaign. This model enables more precise planning of operational procedures by having the operator/planner walk around the virtual environment using a virtual reality headset such as the Oculus rift or the HTC vive. In this virtual environment there is a visual representation of the received biological dose. This can be used to help reduce the doses received by workers by allowing multiple routes to an area of interest to be tried and thus avoiding areas of high dose. This model also provides safe virtual operational training and rehearsal, visualization of radiation dose rates, and estimation of doses received by workers.
Neutronics study on HCCB blanket for CFETR

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Chinese Fusion Engineering Testing Reactor (CFETR) is aimed to obtain the technologies to fill the gaps between ITER and DEMO. The helium cooled ceramic breeder (HCCB) blanket is one of the candidates for CFETR. Ceramics Li₄SiO₄, beryllium and helium of 8 MPa were selected as tritium breeding material, neutron multiplication and coolant, respectively. CLF steel developed in SWIP, one of reduced activation ferritic/martensitic (RAFM) steels, was applied as structural materials of blanket. Neutron and gamma transport was simulated. Activation calculation was performed with FISPACT-2007 code and EAF-2007 activation file. In this study, neutronics characteristic of HCCB blanket such as tritium breeding ratio (TBR), nuclear heating distribution and shielding were assessed. The arrangement of Be, Li₄SiO₄ and cooling plate in blanket module was optimized to improve TBR. The sensitive study for First Wall on TBR was also performed. The radioactivity inventories of HCCB blanket were estimated. The dominant radionuclides in structural materials, Li₄SiO₄ and Beryllium were discussed, respectively. The damage of plasma facing components such as First Wall and divertor due to DT neutron irradiation was analyzed in terms of DPA values.
P4.202

Preliminary risks analysis of the IGNITOR Project realization phase

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In the framework of the joint Russian – Italian collaboration on the development of the IGNITOR project some preliminary estimates of the risk factors that may be occurring during the realization of the project were recently carried out. A distinctive feature of the IGNITOR project is the fact that it contains some innovative solutions in the areas of research, engineering and technology, often having no analogues not only in industry but also outside the specific laboratories and research centers responsible for the development of necessary components. In addition, it is necessary to point out several peculiarities of the IGNITOR project, which distinguish it from other large-scale scientific projects in the sphere of controlled thermonuclear fusion with magnetic confinement, implemented on the basis of the tokamak technology, and which are risk-related in terms of the project realization: 1. The super strong magnetic fields (up to 13 T); 2. The high plasma current discharge (up to 11 MA); 3. Ohmic heating as the main mechanism of ignition of the thermonuclear fusion reaction. During of the risk analysis investigation the following categories of risks were identified: • political; • economical; • achievement of the main goal of the project; • technical and technological risks; • risks of implementation of the scientific research program; • environmental, safety and socio-economical risks. The different impact factors on the realization phase of the IGNITOR project are shown and analyzed. The conclusions of the risks analysis that were obtained are summarized in the joint Table, where the risk category, the description of the problem, circumstances, risk mitigation method and comments are displayed.
Neutronics analysis of a stellarator power reactor based on the HELIAS concept

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The Helical-Axis Advanced Stellarator (HELIAS) is the leading stellarator concept in Europe. Its prime example, Wendelstein 7-X, successfully started operation in 2015. Based on the 5-field-period symmetry, the HELIAS 5-B engineering design study emerged which is a stellarator power reactor concept designed for 3000MW fusion power. The stellarator confines the hot plasma by external field coils only leading to a complex 3D topology of the magnetic configuration. However, the use of specifically shaped non-planar field coils is necessary to generate the helical modulation of the magnetic field. This type of fusion reactor represents a challenging task for the design and maintenance of technological components such as the breeder blanket and the radiation shield. In this context, a major task is related to the neutronic analysis and optimization which must proof the ability of the power reactor to breed the tritium required for self-sufficiency and provide sufficient shielding to protect the super-conducting magnetic field coils. The HELIAS concept was developed at the Max-Planck-Institute for Plasma Physics (IPP) to demonstrate the capability of optimized stellarators to work as fusion power reactors. To this end, the neutronic performance of the HELIAS fusion reactor needs to be assessed. This requires the development of a suitable computational approach to describe the generation of source neutrons in the plasma chamber and to simulate the subsequent particle transport through the complex HELIAS geometry. In this paper, the development of a HELIAS user specific neutron source model for the Monte-Carlo particle transport code MCNP is described. Additionally, an improved CAD model of the HELIAS 5-B engineering design is presented containing breeding blanket, shielding modules, vacuum vessel and magnetic field coils. The CAD model is used to generate a mesh model, which is targeted for the application with MCNP6 for the first neutronic analysis of a HELIAS fusion reactor.
P4.204

**Economic assessment of different operational reactor cycle structures in a pulsed DEMO-like power plant**

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The cost of the electricity (COE) generated by a fusion power plant is a key driver for the technology future energy market deployment. Hence, the ongoing researches on the pulsed DEMO design optimization, taking into account the physical and technical constraints, are putting priorities on the minimization of the DEMO direct costs that indeed greatly influence the COE. Also the duty cycle of a pulsed fusion power plant, defined as the ratio between the burn time and the duration of the operational cycle, has a relevant weight on the COE. In fact maximizing the burn phase fraction and hence the electricity production turns into the COE minimization. However the typical operational cycle of a pulsed fusion power plant consists of a sequence of phases whose length cannot be chosen arbitrary. Researches are currently underway to estimate the optimal duration of each phase as a function of both physical (e.g. plasma stability) and technical (e.g. structural mechanical stresses) constraints. The recommendations arising from the most recent studies on each specific cycle phase are gathered and used to perform sensitivity analysis on the COE. Specifically, a pulsed DEMO-like power plant with a burn time of 2 hours is modeled with the aid of the FRESCO System Code and an assessment of the effect of the operational cycle structure on the COE is carried out. The study is also supported by stochastic analyses. This work is intended to contribute in the economic assessment of a pulsed DEMO-like power plant and support the DEMO design optimization activities.
The Economic Benefits of Big Science R&D Program: With a Focus on KSTAR Program

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This paper is focused on the analysis of spillover benefits of the ongoing R&D program on nuclear fusion in Korea. The spillover effects are understood here as positive externalities of publicly funded R&D activities that may be revealed at the companies’ level in the form of newly created knowledge stock; development of innovative products/processes with broader market applications; strengthening of R&D, manufacturing and marketing capabilities; etc. And this study critically reviews the literature on the economic benefits of publicly funded basic research. In that literature, two main methodological approaches have been adopted—surveys and case studies. These studies have also highlighted the importance of spillovers and the existence of localization effects in research. From the literature based on surveys and on case studies, it is clear that the benefits from public investment in basic research can take a variety of forms. We classify these into seven main categories, reviewing the evidence on the nature and extent of each type. The results demonstrate that KSTAR programs have relatively outstanding performance in seven categories: (1) increasing the stock of useful knowledge; (2) training skilled graduates and researchers; (3) creating new scientific means and methodologies; (4) forming networks and stimulating social interactions; (5) reinforcing the capacity for scientific and technological problem-solving; (6) creating new firms; and (7) access to scientific facilities. In particular, those projects were observed to form an industrial ecosystem for nuclear fusion that extends to the accelerator sector, in the category of creating new firms, while making a significant contribution to training talented researchers and expanding social networks as well. We reconsider the rationale for government funding of basic research, arguing that the traditional ‘market failure’ justification needs to be extended to take account of these different forms of benefit from basic research.
The project & quality management activities in ENEA fusion department

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Ever since the ENEA Fusion Department has been involved in the technology transfer of its knowledge in the field of nuclear fusion from the R&D scope to the execution of large projects together with industry, it has been outlined the importance of working by a quality management system (QMS) and of applying the principles of the Project Management. The head of the ENEA Fusion Department took in 2009 the decision to implement a QMS in accordance with the requirements of EN ISO 9001. The ISO 9001 certification was acquired by ENEA in 2011. The implementation of the QMS led the Fusion Department to apply a series of procedures, not only for contracts where that was specifically requested by customers, but also to all activities of the Department. The main procedures implemented are dealing with the documents and records management, the instrumentation management, the performance of internal audits, the management of non-conformities, the carryout of preventive and corrective actions, the management of personnel and procurement, those devoted to the management of the primary processes (design, development and experimental tests of components and systems for nuclear fusion plants, including construction of related test prototypes), the data analysis aimed at continuous improvement of the effectiveness of the QMS (adoption of indicators). This horizontal approach is very useful when carrying out contracts with F4E, ITER or Eurofusion when it is compulsory to implement the actions described in the quality plans. Actually, most of these actions refer to the application of project management principles. This paper describes in detail the experience gained at ENEA in implementing a QMS in an atypical context such as that of a research organization. The training path carried out by a young researcher hired in the frame of an EUROfusion Grant issued for the purpose is also described.
<table>
<thead>
<tr>
<th>Name</th>
<th>Code</th>
<th>Pages</th>
</tr>
</thead>
<tbody>
<tr>
<td>ABDEDDAIM, Redha</td>
<td></td>
<td>P2.034</td>
</tr>
<tr>
<td>ABDEL MAKSOUD, Walid</td>
<td></td>
<td>O1C.3, P2.089</td>
</tr>
<tr>
<td>ABDEL-MAKSOUD, Walid</td>
<td></td>
<td>P2.087</td>
</tr>
<tr>
<td>ABDRASHTOV, G.</td>
<td></td>
<td>P4.038</td>
</tr>
<tr>
<td>ABE, Ganji</td>
<td></td>
<td>P3.022</td>
</tr>
<tr>
<td>ABELLA, Jordi</td>
<td></td>
<td>P4.165, P4.192</td>
</tr>
<tr>
<td>ABHANGI, Mitul</td>
<td></td>
<td>P3.002</td>
</tr>
<tr>
<td>ABOU-SENA, Ali</td>
<td></td>
<td>P3.118</td>
</tr>
<tr>
<td>ABRAMS, T.</td>
<td></td>
<td>O4B.3</td>
</tr>
<tr>
<td>ACHARD, Joelle</td>
<td></td>
<td>P2.034</td>
</tr>
<tr>
<td>ADAMEK, Jiri</td>
<td></td>
<td>P3.126</td>
</tr>
<tr>
<td>ADEGUN, Joseph</td>
<td></td>
<td>P2.001, P4.018</td>
</tr>
<tr>
<td>AGARICI, Gilbert</td>
<td></td>
<td>P1.025</td>
</tr>
<tr>
<td>AGARWAL, Jyoti</td>
<td></td>
<td>P4.014</td>
</tr>
<tr>
<td>AGEORGES, Nancy</td>
<td></td>
<td>P1.058, P4.055, P4.056</td>
</tr>
<tr>
<td>AGOSTINETTI, Piero</td>
<td></td>
<td>P2.007, P2.021, P2.144, P4.020, P4.021</td>
</tr>
<tr>
<td>AGOSTINI, Pietro</td>
<td></td>
<td>O1A.4</td>
</tr>
<tr>
<td>AGRAWAL, Jyoti</td>
<td></td>
<td>P4.163</td>
</tr>
<tr>
<td>AGUIAM, D.</td>
<td></td>
<td>I4.3</td>
</tr>
<tr>
<td>AGUIAM, Diogo EloI</td>
<td></td>
<td>P4.086</td>
</tr>
<tr>
<td>AHN, Do-Hee</td>
<td></td>
<td>P4.153, P4.154</td>
</tr>
<tr>
<td>AHN, H.J.</td>
<td></td>
<td>P4.104</td>
</tr>
<tr>
<td>AHN, Hee Jae</td>
<td></td>
<td>P2.117</td>
</tr>
<tr>
<td>AHN, Hee-Jae</td>
<td></td>
<td>P4.002, P4.004, P4.006, P4.090</td>
</tr>
<tr>
<td>AHN, Hyeon-Sik</td>
<td></td>
<td>O1B.2</td>
</tr>
<tr>
<td>AHN, Mu-Young</td>
<td></td>
<td>P1.165, P1.189, P1.192, P2.135, P3.165</td>
</tr>
<tr>
<td>AHN, W.-K.</td>
<td></td>
<td>P1.079</td>
</tr>
<tr>
<td>AIBA, Nobuyuki</td>
<td></td>
<td>P1.041</td>
</tr>
<tr>
<td>AIELLO, Gaetano</td>
<td></td>
<td>O5B.3, P1.054, P3.019, P3.026</td>
</tr>
<tr>
<td>AIELLO, Giacomo</td>
<td></td>
<td>P1.146, P1.167, P1.171, P3.117, P3.127</td>
</tr>
<tr>
<td>AINTS, Mart</td>
<td></td>
<td>P4.066</td>
</tr>
<tr>
<td>AJESH, P.</td>
<td></td>
<td>P4.034</td>
</tr>
<tr>
<td>AKERS, Rob</td>
<td></td>
<td>P4.062</td>
</tr>
<tr>
<td>AKERS, Robert</td>
<td></td>
<td>P4.076</td>
</tr>
<tr>
<td>AKIRA, Tonegawa</td>
<td></td>
<td>P4.112</td>
</tr>
<tr>
<td>AKIYOSHI, Masafumi</td>
<td></td>
<td>P3.173</td>
</tr>
<tr>
<td>AKTAA, Jarir</td>
<td></td>
<td>P1.146, P3.131</td>
</tr>
<tr>
<td>ALAM, Md Mahbub</td>
<td></td>
<td>P1.046, P2.078</td>
</tr>
<tr>
<td>ALBAJAR, Ferran</td>
<td></td>
<td>P1.029, P3.024, P3.025</td>
</tr>
<tr>
<td>ALBANESE, R.</td>
<td></td>
<td>I5.3</td>
</tr>
<tr>
<td>ALBANESE, Raffaele</td>
<td></td>
<td>P1.112, P2.005, P4.044</td>
</tr>
<tr>
<td>ALBERRO, Gorka</td>
<td></td>
<td>P4.191</td>
</tr>
<tr>
<td>ALBERTI, Stefano</td>
<td></td>
<td>O5B.3, P3.025, P3.067</td>
</tr>
<tr>
<td>ALDO, Di Zenobio</td>
<td></td>
<td>P2.006</td>
</tr>
<tr>
<td>ALEGRE, Daniel</td>
<td></td>
<td>P4.121</td>
</tr>
<tr>
<td>ALEKSEEV, Andrey</td>
<td></td>
<td>P3.054, P4.051, P4.053</td>
</tr>
<tr>
<td>ALEKSEEV, G. Andrey</td>
<td></td>
<td>P4.055, P4.056</td>
</tr>
<tr>
<td>ALEMAN, Agustin</td>
<td></td>
<td>P2.126</td>
</tr>
<tr>
<td>ALFARO, Luis</td>
<td></td>
<td>P4.063</td>
</tr>
<tr>
<td>ALLELEIN, Hans-Josef</td>
<td></td>
<td>P3.196</td>
</tr>
<tr>
<td>ALLINSON, M.</td>
<td></td>
<td>P4.169</td>
</tr>
<tr>
<td>ALONSO, David</td>
<td></td>
<td>P2.160, P2.161, P2.163</td>
</tr>
<tr>
<td>ALONSO, Javier</td>
<td></td>
<td>O1B.4, O3A.4, P2.018</td>
</tr>
<tr>
<td>ALONSO, Jesus</td>
<td></td>
<td>P2.103</td>
</tr>
<tr>
<td>ALOTTO, Piergiorgio</td>
<td></td>
<td>P2.040</td>
</tr>
<tr>
<td>ALVAREZ, Luis</td>
<td></td>
<td>O1B.4, P2.018</td>
</tr>
<tr>
<td>ALVARO, Elena</td>
<td></td>
<td>P2.101</td>
</tr>
<tr>
<td>ALVES, Eduardo</td>
<td></td>
<td>P3.110</td>
</tr>
<tr>
<td>AMBROSINO, Giuseppe</td>
<td></td>
<td>P4.042, P4.044</td>
</tr>
<tr>
<td>AMBROSINO, Roberto</td>
<td></td>
<td>P1.112, P2.005, P4.019</td>
</tr>
<tr>
<td>AMOSOV, Victor</td>
<td></td>
<td>P4.010, P4.010, P4.011</td>
</tr>
<tr>
<td>AN, Young Hwa</td>
<td></td>
<td>P2.127</td>
</tr>
<tr>
<td>AN, YoungHwa</td>
<td></td>
<td>P4.059</td>
</tr>
<tr>
<td>ANA, George</td>
<td></td>
<td>P4.155, P4.156</td>
</tr>
</tbody>
</table>
ANAND, Rohit P4.034
ANANYEV, Sergey P1.150
ANASHKIN, Igor P1.094
ANCHAROV, Alexey P3.125
ANDA, Gabor P2.057
ANDERSON, J. O2C.3
ANDERSSON SUNDEN, Erik P1.064
ANDERSSON-SUNDEN, Erik P2.053
ANDO, Masami P1.178, P1.180, P1.183
ANDREBE, Yanis P3.028
ANDREENKO, Evgeny P4.051, P4.053
ANDREEVA, Tamara P1.011
ANDREW, Philip P4.058, P4.077
ANDREW, Pivkov P2.133
ANDRULEVICIUS, Mindaugas P3.170
ANEMONA, Alessandro P2.006
ANGELONE, Maurizio P1.066
ANNINO, Carmela P1.138
ANSPOKS, Andris P1.174
ANTHOINE, David P2.048
ANTIPENKO, Alexander P1.131
ANTOHE, Stefan P3.111
ANTONIO, De Lorenzi P1.024
ANTONIO, Masiello P1.024
AOKI, Akira P3.137
APICELLA, Maria Laura O3A.3
APPEL, Lynton P2.042
APPI, Antonio P2.010
APRILE, Daniele P2.020, P2.021
AQUARO, D. P1.140
ARADI, Matyas P2.057
ARAKCHEEV, Aleksey P3.125
ARAKI, Kuniaki P1.046, P2.078
ARDEN, Nils P4.091, P4.093
ARENA, Pietro O1A.4, P1.167, P2.146, P2.152, P3.005, P3.007, P3.117
ARKHIPOV, Alexey P4.077
ARKHIPOV, Igor P3.124
ARMSTRONG, David O2A.1
ARRANZ, Fernando P3.003
ARSHAD, Shakeib P1.048, P3.054
ARSLANOVA, Daria P4.012
ARTAUD, Jean-Francois P4.019, P4.028
ARTS, Karsten P4.124
ARUN PRAKASH, A. P3.015
ASADULIN, Gleb P4.051
ASAKURA, Nobuyuki P1.041, P1.042, P1.115, P3.115, P3.137, P4.120
ASCASIBAR, Enrique P3.175, P4.029
ASDEX UPGRADE TEAM O5C.3
ASDEX UPGRADE TEAM, and the P1.038
ASDEX UPGRADE TEAM, the P4.047, P4.083
ASDEX UPGRADE, Team P1.037, P1.111, P4.123
ASENJO, Jose P3.014
ASH, Andrew P3.077
ASHI KAWA, Naoko P4.115
ASUDANI, Kumudni P3.015
ASZTALOS, O. P1.017
ASZTALOS, Ors P1.074
ATNAFU, Neway P3.089
ATREY, P.K. P3.015
AUBERT, Julien P1.167, P1.171, P3.117
AUMEUNIER, Marie Helene O3A.4
AUMEUNIER, Marie-Helene P1.071
AVEG, Kumar P3.015
AVILES SANTILLANA, Ignacio P1.187
AVOTINA, Liga P3.111
AVRAMIDIS, Konstantinos O5B.3, P1.029
AVRAMIDIS, Kostas P3.025, P4.020
AWANZINO, Cedric P4.054
AYLLON, Juan P1.055
AZUMA, Keisuke P3.114, P3.173
BABINOV, Nikita P4.058
BABU, Gattu R. P3.015
BACHMANN, C. I3.3, P3.128
BADER, Amro P4.020, P4.025
BADER, Michael P1.026, P3.024
BADZIAK, Jan P2.206
BAE, Jinho P1.002, P1.003
BAEZA, Edu P2.204
BAGOT, Paul O2A.1
BAGRETS, Nadezda P1.096
BAILAGI, Nitin P3.015
BAK, Jun-Gyo P2.046
Bakaeva, Anastasia P3.105
BAKLAR, Tomas P1.014
BAKARDJIEVA, Snejana P3.179
BAKLANO, Victor P4.116
BAKLANO, Viktor P1.109
BALASUBRAMANIAM, K. P2.026
BALBA, Itziar O4A.1, P4.061, P4.119
BALDEN, Martin P4.123
BALME, Stephane O3A.4
BALORIN, Colette P1.069, P1.071
BALSHAW, N. P3.048
BALSHAW, Nick P2.003
BALTADOR, Carlo P2.021
BANAUDDA, Moni P3.015
BANDUCH, Martin P1.012
BANDYOPADHYAY, Mainak P2.027, P4.013
BANETTA, Stefano P2.101, P2.102, P2.185
BANETTA, Steffano P2.103
BANG, Eunnam P2.116
BANKS, J. P4.169
BANKS, Joe P4.170
BANSAI, Gourab P4.013
BANSAL, Laxmikant K. P3.068
BAO, Hui P2.155
BAOGUO, Pan P4.193
BARABASCHI, P. I1.3, I3.2
BARBATO, Lucio P1.112
BARBERO SOTO, Jose Luis P1.125
BARILLAS, Laura P3.014
BARNESLEY, Robin P4.050, P4.059
BARONE, Gianluca P2.013, P4.190
BAROSS, Teteny P1.125, P2.186, P3.053
BARRACHIN, Marc O4C.4
BARRERA, Eduardo P3.060
BARRETT, Thomas R. O2B.1, P1.123
BARRETT, Tom P3.159
BARRETT, Tom R P1.112
BARTKOVA, Denisa P3.191
BARTON, Justin E. O5C.4, P3.042
BARUAH, U.K. O1C.2
BARUAH, Ujjwal P3.062
BARUAH, Ujjwal Kumar P3.068
BAS, Isidro P1.048
BASSAN, Michele P4.058
<table>
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<th>Name</th>
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<td>P2.057, P2.061</td>
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<td>P1.055, P2.048, P2.127, P3.203, P4.059</td>
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<td>BERINNETTI, Andrea</td>
<td>P1.029</td>
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<td>BERTIZZOLO, Robert</td>
<td>P3.020, P3.021</td>
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<td>BESSETTE, Denis</td>
<td>O1C.1</td>
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<td>P3.087</td>
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<td>O1B.3, P1.075, P2.038, P2.039, P2.040, P2.041, P4.045</td>
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<td>P1.070</td>
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<td>BHARATHI, Punjapryu</td>
<td>P3.068</td>
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<td>BHATT, K. S.</td>
<td>P1.001</td>
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<td>BHATT, Shailesh</td>
<td>P3.143</td>
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<td>BHOOMI S, Gajjar</td>
<td>P1.186</td>
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<td>BIANCHI, Aldo</td>
<td>P4.202</td>
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<td>BIEDERMANN, Christoph</td>
<td>P2.064, P2.068, P2.070</td>
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<td>BIEG, B.</td>
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<td>O1B.1, P4.003, P4.003, P4.020</td>
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<td>P1.021, P1.025</td>
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<td>P4.036, P4.037</td>
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<td>P2.123, P2.124, P2.125</td>
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<td>O1B.4, P2.018</td>
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<td>P1.167, P3.117</td>
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<td>P1.131, P4.162</td>
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<td>P3.047, P3.048</td>
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<td>BRAIC, Viorel</td>
<td>P3.046</td>
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<td>BRANAS, Beatriz</td>
<td>P3.003, P4.185</td>
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<td>BRANDT, Christian</td>
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<td>BRAUNMUELLER, Falk</td>
<td>O5B.3, P3.025</td>
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<td>BREDAR, Mauro</td>
<td>P1.025, P4.009</td>
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<td>BREMOND, Sylvain</td>
<td>P1.070, P4.042, P4.043</td>
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<td>BREZINSEK, Sebastijan</td>
<td>P3.034, P4.121</td>
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<td>BRIANI, Pierfederico</td>
<td>P2.126</td>
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<td>P2.053</td>
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<td>P4.054</td>
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<td>BRIESEMEISTER, A.</td>
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<td>BRIGHENTI, Alberto</td>
<td>P1.084</td>
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<td>BRINKMANN, Jens</td>
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<td>BRIX, Matias</td>
<td>P3.049</td>
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<td>BROCCA, Claudio</td>
<td>P2.023</td>
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<td>BROECKMANN, Christoph</td>
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<td>BROLATTI, Giorgio</td>
<td>P2.051</td>
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<td>BROMBIN, Matteo</td>
<td>P2.020, P2.079</td>
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<td>BROOKS, Arthur</td>
<td>P3.152</td>
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<td>BROOKS, J.N.</td>
<td>O4B.3</td>
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<td>BROSLAWSKI, Andrzej</td>
<td>P3.045, P3.047</td>
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<td>BROSZAT, Torsten</td>
<td>P2.071</td>
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<td>BROWN, Richard</td>
<td>P3.198</td>
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<td>BROWN, Thomas</td>
<td>P2.167</td>
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<td>BRUN, Emeric</td>
<td>O5A.3</td>
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<td>BRUNI, Giacomo</td>
<td>P4.149</td>
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<td>BRUNKHORST, Christopher</td>
<td>P3.035</td>
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<td>BRUNO, Vincent</td>
<td>O2C.1, P1.144</td>
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<td>BRUSCHI, Alessandro</td>
<td>P2.007, P3.025</td>
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<td>BRUSCHI, Alex</td>
<td>O5B.3</td>
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<td>BRUZZONE, Pierluigi</td>
<td>O1C.1, O1C.4, P1.085, P1.086, P1.087, P1.089</td>
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<td>BUCALOSSI, J.</td>
<td>I5.2, P4.101</td>
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<td>BUCALOSSI, Jerome</td>
<td>O2B.3, P1.069, P1.070, P1.136, P3.113, P3.123</td>
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<td>BUDAEV, Viacheslav</td>
<td>P3.124</td>
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<td>BUFFA, Fabrizio</td>
<td>P4.009</td>
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<td>BUELER, Leo</td>
<td>P1.168, P2.151</td>
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<td>BUKREEV, Ivan</td>
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<td>BULT, David</td>
<td>P3.018</td>
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<td>BUNTING, Patrick</td>
<td>P4.119</td>
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<td>BUONGIOVI, Gaetano</td>
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<td>CALDERONI, Patrick</td>
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<td>CALVO PLAZA, Francisco Jose</td>
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<td>CAMP, P.</td>
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CAMP, Patrick P2.003
CAMPAGNOLO, Roberto P1.048
CAMPLING, D. I5.5
CANAS, Daniel P2.017
CANDIDO, Luigi P4.166
CANDURA, Donatella O4B.2
CANTON, Alessandra O1B.3
CANTONE, Bruno O3A.4
CAO, Chengzhili P1.005
CAO, Chenzhi P4.157
CAPEL, Adrian P4.061
CAPOBIANCO, Roberto P3.069
CAPPA, Alvaro P3.075
CARDELLA, A. O1B.4
CARDELLA, Antonino P1.138, P2.018
CARDENES, Sabas P2.101
CARDINALI, Alessandro P2.007
CARELLA, Elisabetta O1A.1, P1.155, P2.162
CARLIN, Yann P1.146
CARLINI, Maurizio P4.150
CARLONI, Dario P1.145, P3.197
CARLS, Andre O5C.1, P1.013, P2.063
CARMONA, Jose Miquel P1.049
CARNEVALE, Daniele P3.043
CARPIGNANO, Andrea P3.197
CARRASCO, Ricardo P4.029
CARR, Gabriel P2.182
CARROZZA, Saverio P1.021
CARUSO, Gianfranco O1A.4, P2.149, P2.150
CARVALHO, Bernardo O4A.3, P1.048, P3.065
CARVALHO, Bernardo B. P3.063, P3.206
CARVALHO, I.S. P4.169
CARVALHO, Ivo P4.170
CARVALHO, Ivo S. P1.045
CARVALHO, P.F. P2.050
CARVALHO, Paulo O4A.3, P3.064, P3.065, P4.135
CARVALHO, Paulo F. P3.063, P3.206, P4.086
CARVALHO, Pedro J. P4.086
CASAL, Natalia P3.056, P3.057, P3.070, P3.203
CASTANO BARDAWIL, David Antonio P4.060
CASTELLANOS, Jesus O1A.1, P2.162, P3.008
CATEG, J. P. I3.3
CATEG, Juan Pablo P2.004, P3.203
CATEG, Juan-Pablo P2.142
CAU, Francesca P1.029, P2.126, P3.085
CAUGHMAN, John P4.113
CAVALIER, Jordan P2.058, P3.126
CAVAZZANA, Roberto O1B.3, P1.075, P2.040, P4.086
CAVE-AYLAND, K. P3.077
CECCONELLO, M. P2.050
CECCONELLO, Marco P4.041
CECCUZZI, Silvio P2.007, P4.024
CENEDELLA, Gabriele P1.025
CENGHER, Mirela P2.037
CENTIOLI, C. P2.050
CENTIOLI, Cristina P3.044, P3.066, P4.088
CERACCHI, Andrea P4.008
CERRI, Valerio O4B.2
CERVARO, Vannino P2.079
CHAI, Zhiyong P1.187
CHAKRABORTY, A. I4.2
CHAKRABORTY, Arun P2.026, P4.013
CHAKRABORTY, Arunkumar P2.027
CHAKRAPANI, Chodimella P3.068
CHALLIS, Clive P1.045
CHANG, Doo-Hee P4.026, P4.027
CHANG, Han Soo P4.205
CHANG, Xiaobo P1.123
CHANG, Yong Bok P4.001
CHANG, Yong-Bok P4.002, P4.004
CHANG, Yoon-Suk P2.134
CHAPPUIS, Philippe P2.101, P2.130, P4.126
CHAUDHARI, Bhum P3.015
CHAUDHARI, Vishnu P1.069, P1.070
CHAUHAN, Pradeep K. P3.015
CHAUIMEIX, Nabiha P3.196
CHAVIN, Didier P2.065
CHAVDA, Chhaya P1.070
CHELIS, Ioannis O5B.3, P3.025
CHEN, Changan P3.162
CHEN, Chao P2.203
CHEN, Chao bin P2.016
CHEN, Chenyuan P4.157
CHEN, Gangyu P1.032
CHEN, Hongli P1.158, P1.159, P1.163, P2.165, P3.163, P4.084
CHEN, Jiming P2.187
CHEN, Lei P4.145
CHEN, Qian P2.138
CHEN, R P1.067
CHEN, Shanqi P3.193
CHEN, Tan P1.142
CHEN, Y. I4.3
CHEN, Yixue P2.157
CHEN, Yuhua P4.145
CHEN, Yue P4.159
CHEN, Yuming O3B.2
CHEN, Zhibin P2.203
CHENG, Mengyun P2.016
CHENG, Xiaoman P2.154, P2.156, P3.201
CHENG, Yong P1.144, P4.085
CHEON, Mun Seong P2.127
CHEON, MunSeong P4.059
CHERNAKOV, Alexander P4.058
CHERNAKOV, Anton P4.058
CHERNYSHOVA, Maryna P3.073, P4.068, P4.070, P4.071
CHIARIELLO, Andrea Gaetano P1.044, P2.039
CHIKHRAIY, Evgeniy P4.116
CHIKHRAIY, Yevgen P1.109
CHIOVARO, Pierluigi P2.146, P2.152
CHIRON, Jacques P3.086, P3.087
CHITARIN, Giuseppe P2.020, P2.021
CHMIAA, Mohamed P2.034
CHO, Hyoung-Kyu P4.159
CHO, Jungyul P3.202
CHO, K. P1.079
CHO, Moohyun P2.032, P2.033
CHO, Seungyon P1.165, P1.166, P1.188, P1.189, P1.192, P3.150, P3.165, P4.106
CHOI, Wook P4.005
CHOI, C. H. P1.184
CHOI, Chang-Ho P1.127
CHOI, Chang-Ho Choi P1.130
CHOI, Chang-Hwan O3C.1
CHOI, Dae Jun P4.005
COSTA, Pietro P2.011
COSTEA, Stefan P4.069
COURTOIS, Xavier P1.070, P1.071, P3.113
COUTURIER, Steve P3.067
CRACIUNESCU, T. P3.047, P3.048
CRACIUNESCU, Teddy P3.046
CRAMP, Simon O4A.1
CREPEL, Bruno P2.091
CRESCENZI, Fabio P1.107, P1.117
CRISANTI, Flavio P2.005, P2.006, P2.010
CRISTESCU, Ion P4.156
CROCI, Gabriele P2.080
CROFT, D. P3.047, P3.048
CROFT, David P2.004, P3.046
CROFTS, Oliver P3.140
CROWLEY, B. P2.035
CROWLEY, Brendan P2.036
CRUZ MALAGON, Dario Andres P4.017
CRUZ, Nuno O4A.3, P2.050, P3.063, P3.066
CSEH, Gabor P2.068
CUANILLON, Philippe P3.067
CUCCIARO, Antonio P2.086, P2.088
CUFAR, A. P1.066, P3.048
CUI, Hu P3.114
CUNNINGHAM, Geoffrey P2.047
CURSON, Paul P1.040
CURUIA, M. P3.047, P3.048
CURUIA, Marian P3.046
CZARNECKA, Agata P2.070
CZARSKI, Tomasz P3.073, P4.068, P4.070, P4.071
D’AMICO, Gabriele P2.101
D’ELIA, Gerardo P4.158
DAIRAKU, Masayuki P2.028
DAL BELLO, Samuele O1B.3, P2.022, P4.009
DALIN, Jean-Michel P1.187
DALLA PALMA, Mauro O1B.3, P2.022, P2.079
DAMBA, Julius P4.029
DANANI, C. P1.092
DANG, Tongqiang P2.016
DANGI, Jinendra P1.006
DANIEL, Raju P1.070
DANILOV, Igor P2.130
DARANYI, Fruzsina P3.053
DARBOS, Caroline P3.023, P3.024
DAS, Amita P3.015
DAVE, R. O1C.2
DAVE, Rasesh P3.062
DAVIS, Sam P2.086, P2.091
DAVYDENKO, Vladimir I. P3.030
DAY, Chr. P3.128
DAY, Christian O5A.1, P2.022, P3.012, P4.161, P4.164
DAY, Ian E P2.002
DE ANGELI, Marco P3.126
DE ARCAS, Guillermo P3.060
DE BLAS, Alfredo P2.204
DE CASTRO, Alfonso P4.121
DE CASTRO, Vanessa P1.174
DE GENTILE, Benoît O3A.4
DE LA CAL, Eduardo O3A.4, P4.054
DE MAGISTRIS, Massimiliano P4.044
DE MARNE, Pascal P4.083
DE MASI, Gianluca P4.086
DE MEIS, Domenico P3.131
DE MURI, Michela P2.079, P4.007
DE PRADO, Javier P2.188
DE TEMMERMAN, Gregory P1.058, P3.051, P4.121
DE TOMMASI, Gianmaria P4.042, P4.043
DE VRIES, Peter P4.044
DE VRIES, Peter Christiaan P4.046
DE WIT, Shaun P4.062, P4.076
DEAN, Buchenauer P3.114
DE CAMPS, Hans P1.021, P1.022, P2.023, P3.082, P3.083
DECANIS, Christelle P2.017
DECOOL, Patrick O1C.3, P2.087, P2.089, P2.091
DEGLI AGOSTINI, Fabio P2.144
DEGRASSIE, J. O2C.3
DEL NEVO, Alessandro O1A.4, P2.146, P2.147, P2.148, P2.149, P2.150, P2.152, P2.153, P3.135
DEL SERRA, D. P1.140
DELAPLANCHE, Jean-Marc O2C.1
DELCHAMBRE, Elise O3A.4
DELIN, Chu P4.193
DELLA CORTE, Antonio P1.086, P2.006
DELLABIANCIA, Mattia P2.108
DELOPOULOS, Georges P2.101
DELMAS, Etienne O2C.1
DELOGU, R.S. P4.039
DELPECH, Lena P2.032
DEMANGE, David O1A.1, P4.151
DEMBKOWSKA, Aleksandra P1.083
DEN HARDER, N. I2.1
DENG, Weiping P2.158
DESGRANGES, Corinne O2B.3
DESHENG, Cheng P4.193
DEVRED, A. I3.1
DEVRED, Arnaud O1C.1
DHANANI, Kalpeshkumar R. P3.015
DHARD, C.P. P1.009
DHARD, Chandra Prakash P1.010
DHOLA, H. O1C.2
DHOLA, Hiteshkumar P3.062
DHONGDE, Jasraj P1.070
DHONGDE, Jasraj R. P3.015
DI FONZO, Fabio P3.147
DI GIRONIMO, Giuseppe O1A.4, P2.009, P2.141, P3.131, P3.132, P3.135
DI MAIO, Pietro Alessandro O1A.4, P1.119, P1.120, P1.167, P2.146, P2.152, P3.005, P3.007
DI PACE, Luigi P4.206
DI PIAZZA, Ivan P2.148, P2.150
DI PIETRO, E. I1.3
DI PIETRO, Enrico P1.138, P2.091
DIAZ, Ester P4.151
DIBON, Mathias P4.080
DIEGLE, Eberhard I4.1
DIES, Javier P2.204
DIMITROVA, Miglena P3.126
DINESCU, Gheorghe P3.108
DING, Fang O2B.4
DING, Kaizhong P4.085
DING, R. O4B.3
DINKLAGE, A. I5.4
DIRK, Wunderlich P2.080
DITTMAR, Timo P4.121
DMITRIEV, Artem P4.058
DNESTROVSKIY, Alexey P4.022
DOBIAS, Petr P2.047
DOCEUL, Louis P1.137, P3.123
DOERNER, R.P. O4B.3
DOFEK, Ivan P1.169
<table>
<thead>
<tr>
<th>Name</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>DOLENSKY, Bernhard</td>
<td>P.4.179</td>
</tr>
<tr>
<td>DOLIZY, Frederic</td>
<td>P.3.030</td>
</tr>
<tr>
<td>DOMALAPALLY, Phani</td>
<td>P.2.108</td>
</tr>
<tr>
<td>DOMPTAIL, F.</td>
<td>P.3.077</td>
</tr>
<tr>
<td>DOMPTAIL, Fred</td>
<td>O2B.1</td>
</tr>
<tr>
<td>DONGIOVANNI, Danilo</td>
<td>P.2.199</td>
</tr>
<tr>
<td>DONNELLY, Stephen E.</td>
<td>P.3.107</td>
</tr>
<tr>
<td>DORIA, Andrea</td>
<td>P.4.024</td>
</tr>
<tr>
<td>DORMIDO-CANTO, Sebastian</td>
<td>P.4.063</td>
</tr>
<tr>
<td>DORN, Christopher</td>
<td>P.4.114</td>
</tr>
<tr>
<td>DORRONSORO, Ander</td>
<td>P.2.081</td>
</tr>
<tr>
<td>DOSTAL, Vaclav</td>
<td>P.2.197, P.2.200</td>
</tr>
<tr>
<td>DOYEN, Olivier</td>
<td>O5A.4</td>
</tr>
<tr>
<td>DRAGHIA, Mirela</td>
<td>P.4.155, P.4.156</td>
</tr>
<tr>
<td>DRAGO, Giovanni</td>
<td>P.2.088</td>
</tr>
<tr>
<td>DRAKSLER, Martin</td>
<td>P.2.140</td>
</tr>
<tr>
<td>DRANITCHNIKOV, Aleksandr N.</td>
<td>P.3.030</td>
</tr>
<tr>
<td>DREMER, Matthias</td>
<td>P.1.131</td>
</tr>
<tr>
<td>DREBEL, Matthias</td>
<td>P.4.162</td>
</tr>
<tr>
<td>DRENK, Aleksander</td>
<td>P.4.121</td>
</tr>
<tr>
<td>DREVON, Jean-Marc</td>
<td>P.3.057, P.4.054</td>
</tr>
<tr>
<td>DREWS, Philipp</td>
<td>P.2.073</td>
</tr>
<tr>
<td>DROZDOWICZ, K.</td>
<td>P.1.196</td>
</tr>
<tr>
<td>DROZDOWICZ, Krzysztof</td>
<td>P.2.053</td>
</tr>
<tr>
<td>DU, Liang</td>
<td>P.1.142</td>
</tr>
<tr>
<td>DUAN, X.R.</td>
<td>P.2.076</td>
</tr>
<tr>
<td>DUAN, Xuru</td>
<td>P.4.095</td>
</tr>
<tr>
<td>DUBAN, Richard</td>
<td>P.2.047</td>
</tr>
<tr>
<td>DUBINKO, Andrii</td>
<td>P.3.105</td>
</tr>
<tr>
<td>DUBROV, Maksim</td>
<td>P.2.045</td>
</tr>
<tr>
<td>DUCHATEAU, Jean-Luc</td>
<td>P.2.087</td>
</tr>
<tr>
<td>DUCKWORTH, Robert</td>
<td>P.4.113</td>
</tr>
<tr>
<td>DUFOUR, Thihaault</td>
<td>P.2.065</td>
</tr>
<tr>
<td>DUMORTIER, Pierre</td>
<td>P.4.036, P.4.037</td>
</tr>
<tr>
<td>DUNAEVSKY, A.</td>
<td>P.4.038</td>
</tr>
<tr>
<td>DUNAI, D.</td>
<td>P.1.017, P.1.074</td>
</tr>
<tr>
<td>DUNAI, Daniel</td>
<td>P.2.057</td>
</tr>
<tr>
<td>DURAN, Ivan</td>
<td>P.1.049, P.1.050, P.1.051, P.2.001, P.4.073</td>
</tr>
<tr>
<td>DURODIE, Frederic</td>
<td>P.1.034, P.3.034, P.4.036</td>
</tr>
<tr>
<td>DURODIE, Frederic</td>
<td>P.4.037</td>
</tr>
<tr>
<td>DUTTA, P.</td>
<td>P.1.092</td>
</tr>
<tr>
<td>DUTTA, Pramit</td>
<td>P.4.130</td>
</tr>
<tr>
<td>DUTTA, S.</td>
<td>P.1.092</td>
</tr>
<tr>
<td>DUVAL, Basil</td>
<td>P.3.027</td>
</tr>
<tr>
<td>DUVAL, Basil P.</td>
<td>P.3.030</td>
</tr>
<tr>
<td>D'ARCANGELO, O.</td>
<td>I.4.3</td>
</tr>
<tr>
<td>D'ARCANGELO, Ocleto</td>
<td>P.4.086</td>
</tr>
<tr>
<td>EADE, T.</td>
<td>I.3.3</td>
</tr>
<tr>
<td>EADE, Tim</td>
<td>P.1.201, P.3.159</td>
</tr>
<tr>
<td>EATON, Russel</td>
<td>P.2.104</td>
</tr>
<tr>
<td>EATON, Russell</td>
<td>P.2.106</td>
</tr>
<tr>
<td>EBELING, Rob</td>
<td>P.4.079</td>
</tr>
<tr>
<td>EBOLE, Marica</td>
<td>O1A.4, P.2.153</td>
</tr>
<tr>
<td>ECHEBEERRIA, Jon</td>
<td>O3B.3</td>
</tr>
<tr>
<td>ECKARDT, Christian</td>
<td>P.2.027</td>
</tr>
<tr>
<td>EDAO, Yuki</td>
<td>P.4.152, P.4.168</td>
</tr>
<tr>
<td>EDLINGTON, Trevor</td>
<td>P.1.125, P.3.053</td>
</tr>
<tr>
<td>EDWARDS, Dan J.</td>
<td>O3B.4</td>
</tr>
<tr>
<td>EDWARDS, Paul</td>
<td>P.4.126</td>
</tr>
<tr>
<td>EGOROV, Konstantin</td>
<td>P.2.063, P.2.106, P.2.130</td>
</tr>
<tr>
<td>EGUIA, Joan</td>
<td>P.2.103, P.2.128</td>
</tr>
<tr>
<td>EHRKE, Gunnar</td>
<td>P.2.123</td>
</tr>
<tr>
<td>EIXENBERGER, Horst</td>
<td>P.4.091, P.4.093</td>
</tr>
</tbody>
</table>
FELTON, R. P4.169
FELTON, Robert P1.045, P2.003, P4.043, P4.170
FENG, Changle P1.121
FENG, Chunhua P4.097
FENG, Jiabo P1.142
FENG, Kaiming P4.201
FENG, S.Y. P1.067
FERLAY, Fabien O2C.1, P3.123, P4.137
FERLET, Marc O3A.4, P4.054
FERNANDES, A. P2.050, P3.046, P3.047, P3.048
FERNANDES, Ana O4A.3, P3.045, P3.066, P4.086
FERNANDEZ, Ivan P2.160, P2.162, P3.119, P3.151
FERNANDEZ, Pilar O1B.4, P2.018
FERNANDEZ-BERCERUELO, Ivan P2.161, P2.163
FERRO, Alberto P2.083, P3.082
FERRO, Giuseppe P3.043
FERRON, John R. O5C.4
FIAMOZZI ZIGNANI, Chiara P2.006
FIETZ, Walter H. P1.096
FIGACZ, Waldemar P2.066, P2.069
FIGINI, Lorenzo P1.033, P1.043, P2.007
FIGUEIREDO, Antonio P1.138
FIGUEIREDO, J. P3.048
FIGUEIREDO, Joao P3.045, P3.046
FIL, Nicolas P1.027
FINCATO, Michele P4.009
FINOTTI, Claudio P2.038, P2.041, P3.083
FIRDAOUISS, Mehdi O2B.3, P1.069, P1.112
FISCHER, U. I3.3
FISHER, B. O2C.3
FLAMMINI, Davide P1.128, P2.004, P2.051
FLAMMINI, Michael P2.131
FLOWER, P. P4.169
FOR THE W7-X TEAM I5.1
FORIS, Laurent O5A.4, P1.146
FORNAGLIONE, Nicola P2.013, P2.014, P2.148, P2.153
FORMISANO, Alessandro P2.039, P4.044
FORMES, Tomasz P2.066, P2.069, P2.070
FORTUNA, Luigi O3A.3
FORTUNEA, Elzbieta P3.049
FORTUNATO, Joao O4A.3
FOSSEN, Arnaud P4.157
FRADELY, Jorge P2.126
FRANCES, Laetitia P4.151
FRANCHIN, Luca P2.079
FRANCK, Joachim O5B.3
FRANKE, Thomas O5B.3, P4.020
FRANZ, Paolo P2.038, P2.071
FRANZA, Fabrizio P3.130, P3.199
FRASCA, Mattia O3A.3
FRASCATI, Fabrizio P3.005
FRATTOILLO, A. O5A.1
FRATTOILLO, Antonio P3.012, P4.148, P4.158, P4.161
FREDERIEC, Cousin P4.195
FREI, Marcel P3.024
FREISINGER, Michael P4.067
FRICKE, Marko P2.093, P2.094
FRICONNEAU, Jean-Pierre P4.131
GOHIL, Gumansinh P1.006
GOLDSTEIN, Igor P1.038
GOLOBOROD’KO, V. P3.046
GOLOBORODKO, V. P3.047
GOLONOROD’KO, V. P3.048
GOMEZ, Alvaro P2.018
GOMEZ, Gerard Escudero P3.083
GONCALVES, B. P2.050
GONCALVES, Bruno O4A.2, O4A.3, P1.048, P3.063, P3.064, P3.065, P3.066, P4.086
GONCALVES, Bruno S. P3.206
GONCHAROV, Pavel P4.022
GONDE, Rene P2.091
GONG, S.B. P2.076
GONICHE, Marc O5B.2, P2.034
GONZALEZ, Jorge P1.125
GONZALEZ, Maria P1.154, P1.155, P2.160, P3.149
GONZALEZ, Miguel P1.049, P3.056, P3.070
GONZALEZ-MARTIN, Javier P1.055
GOODMAN, Tim P3.029
GOODMAN, Timothy P1.033, P3.018, P3.019, P3.020, P3.021
GOODYEAR, Alex P1.045
GORAYEV, G. P2.173
GORBUNOV, Alexey P4.050
GORDEEV, Sergiy P4.172, P4.174
GORDIENKO, Yuri P4.116
GORDIENKO, Yuriy P1.109
GORELOV, Yuri P2.037
GORINI, G. P3.047
GORINI, Giuseppe P3.046
GORNIKEL, Ilya P4.010, P4.010, P4.011, P4.012
GOTO, Takuya P2.172, P3.144, P4.144
GOULDING, Richard P4.037
GOULDING, Richard Howell P4.036
GOULDING, Rick P4.113
GOWLAND, R. P4.169
GRACEFFA, Joseph P4.008
GRAHAM, Bill P2.003
GRAHAM, Margaret P4.037
GRAHAM, Margaret P4.036
GRAHL, Michael P1.011, P2.062, P2.063, P3.036, P3.037
GRANDO, Luca O1B.3, P1.075, P2.040, P2.041, P4.009
GRANUCCI, Gustavo O5B.3, P2.007, P4.019, P4.020
GRANZOTTO, Nicola P1.025
GRASHIN, Sergey P3.124
GRATTAROLA, Marco P1.197
GRAVANTI, Filippo P4.158
GRAVES, Jonathan P. P1.045
GRAVES, Van P4.113
GREAVES, Graeme P3.107
GRECHI, Henri O4B.1, P1.104, P2.125, P2.177, P4.123
GRIBOV, Yuri P4.044, P4.046
GRIFFO, Andrew O4C.3, P2.201
GRIGORIEV, Eduard P4.066
GRIGORIEV, Sergey P3.086, P3.088
GRIFF, Sverker P3.046
GRISHAM, Larry R O2C.2
GROCHOL, Friedrich P4.174
GRONER, Frank
GROS, Gilles
GROSCHEL, Friedrich
GROSSER, Klaus
GROSSETTI, Giovanni
GRULKE, Olaf
GRUN, Martin
GRYAZNEVICH, Mikhail
GRYAZNEVICH, Mikhail P.
GRZONKA, Justyna
GUAN, Wenhai
GUARDI, Filippo
GUASP, Jose
GUILHEM, D.
GUILLEMAUT, C.
GUILLEMAUT, Christophe
GUILLO, Christophe
GUIMARAIS, Luis
GUION, Andrea
GUO, H.Y.
GUO, Yong
GUO, Yun
GUPTA, Chet Narayan
GUPTA, D. K.
GUPTA, Dinesh
GUPTA, Laxminarayan
GUPTA, Neelam
GUPTA, V.
GUTIERREZ, Daniel
GUTRUF, Sven
GWON, Hyoseong
GYERGYES, Tomaz
HA, Min-Su
HA, Minsu
HAAS, Guenter
HACEK, Pavel
HADRABA, Hynek
HAIH, Sang-Hee
HAIH, Sanghee
HAIBING, Wang
HAIBING, Jiang
HAJDUK, Leszek
HAJEK, Petr
HAJNAL, Nandor
HAKOLA, Antti
HAKOLA, Antti
HALITOVS, Mihails
HALL, Stephanie
HALLEN, Anders
HAMADA, K.
HAMAGUCHI, Dai
HAMAGUCHI, Kohei
HAMAGUCHI, Shinji
HAMAJI, Yukinori
HAMANO, Takashi
HAMILTON, David
HAMLYN-HARRIS, C.
HAN, Gi Young
HAN, Jongwon
HANADA, Kazuaki
HERRMANN, Albrecht
HEUER, Simon
HEUSER, Julia M.
HIDALGO, Carlos
HIGAKI, Haruhiro
HIGASHI, Kei
HIGASHIJIMA, Aki
HILLARET, Julien
HINKS, Jonathan A.
HINOKI, Tatsuya
HIRAI, Takeshi
HIRAKAWA, Yasushi
HIRATSUKA, Junichi
HIROE, Fujita
HIROSE, Akira
HIROSE, Takanori
HIROSHI, Kasahara
HISAKA, Chiaki
HISHINUMA, Yoshimitsu
HIWATARI, Ryoji
HJALMARSSON, Anders
HL-2A, Team
HOANG, Tuong
HOASHI, Eiji
HOASHI, Eji
HODGSON, Eric
HODGSON, Eric R.
HODGSON, Eric Richard
HOFFMANN, Jan
HOGGE, Jean-Philippe
HOLCOMB, Christopher
HOLLFELD, Klaus-Peter
HOLLFELD, Klaus-Peter
HOLLINGSWORTH, A.
HOLLOCOMBE, Jonathan
HOLMES, A. J. T.
HOLTZ, Andreas
HOMMA, Yuki
HON, Alexander
HONG, Bong Guen
HONG, Jaesic
HONG, Kwen-Hee
HONG, Suk-Ho
HONG, Suk-ho
HONZU, Toshihiko
HOPF, Christian
HOPPER, Dave
HORACEK, Jan
HORII, Hiroyuki
HORIIKE, Hiroshi
HORIKOSHI, Seira
HORNUNG, Gregoire
HORSTENSMEYER, Y.
HOSAKA, Kazuki
HOSCHEM, Till
HOSEA, Joel
HOSHINO, Kazuo
HOSHINO, Tsuyoshi
HOSINOSI, Tsuyoshi
HOTCHIN, Simon
HOU, Y.M.
HOUBEN, Anne
HOURY, Michael
HROMADKA, Jakub
<table>
<thead>
<tr>
<th>Name</th>
<th>Pages</th>
</tr>
</thead>
<tbody>
<tr>
<td>HRON, Martin</td>
<td>P2.061</td>
</tr>
<tr>
<td>HU, Cui</td>
<td>P3.187, P3.188</td>
</tr>
<tr>
<td>HU, G.H.</td>
<td>P1.017</td>
</tr>
<tr>
<td>HU, Jiansheng</td>
<td>O2B.4, P4.159</td>
</tr>
<tr>
<td>HU, Lijin</td>
<td>P2.015, P2.016</td>
</tr>
<tr>
<td>HUAJUN, Li</td>
<td>P4.094</td>
</tr>
<tr>
<td>HUANG, Bill</td>
<td>P1.095</td>
</tr>
<tr>
<td>HUANG, Hua</td>
<td>P3.009</td>
</tr>
<tr>
<td>HUANG, Kai</td>
<td>P2.154, P2.156, P3.201</td>
</tr>
<tr>
<td>HUANG, Mei</td>
<td>P1.032</td>
</tr>
<tr>
<td>HUANG, Qunying</td>
<td>P1.172, P1.194, P1.195</td>
</tr>
<tr>
<td>HUANG, Shenghong</td>
<td>P4.107</td>
</tr>
<tr>
<td>HUANG, Wangli</td>
<td>P2.202</td>
</tr>
<tr>
<td>HUANG, Xiangmei</td>
<td>P4.157</td>
</tr>
<tr>
<td>HUANG, Yao</td>
<td>P2.042</td>
</tr>
<tr>
<td>HUANG, Yawei</td>
<td>O1C.3, P2.090</td>
</tr>
<tr>
<td>HUANG, Yuncong</td>
<td>P2.138</td>
</tr>
<tr>
<td>HUANG, Zhiyong</td>
<td>P3.162</td>
</tr>
<tr>
<td>HUBER, A.</td>
<td>I2.1</td>
</tr>
<tr>
<td>HUBER, Alexander</td>
<td>O4A.1, P4.061, P4.067, P4.119</td>
</tr>
<tr>
<td>HUBER, Valentina</td>
<td>O4A.1, P4.061</td>
</tr>
<tr>
<td>HUGOT, Francois-Xavier</td>
<td>O5A.3</td>
</tr>
<tr>
<td>HUH, Sung-Ryul</td>
<td>P4.027</td>
</tr>
<tr>
<td>HUILING, Wei</td>
<td>P4.030</td>
</tr>
<tr>
<td>HUMPHREYS, Dave A.</td>
<td>P4.042</td>
</tr>
<tr>
<td>HUMPHREYS, David</td>
<td>P4.043</td>
</tr>
<tr>
<td>HUMPHREYS, David A.</td>
<td>O5C.4</td>
</tr>
<tr>
<td>HUMPHRY-BAKER, Samuel A.</td>
<td>P3.107</td>
</tr>
<tr>
<td>HUMRICKHOUSE, Paul</td>
<td>O4C.3, P1.151</td>
</tr>
<tr>
<td>HUSAK, Roman</td>
<td>P3.190, P3.191</td>
</tr>
<tr>
<td>HWANG, Y.</td>
<td>P1.079</td>
</tr>
<tr>
<td>HWANG, Y. S.</td>
<td>P1.077, P2.043</td>
</tr>
<tr>
<td>HWANG, Y.S.</td>
<td>P1.078</td>
</tr>
<tr>
<td>HWANG, Yongseok</td>
<td>P1.080</td>
</tr>
<tr>
<td>IADICICCO, Daniele</td>
<td>P3.147</td>
</tr>
<tr>
<td>IANNONE, Francesco</td>
<td>P4.158</td>
</tr>
<tr>
<td>IBANO, Kenzo</td>
<td>P1.110, P3.101, P3.102, P3.103, P3.153, P4.110</td>
</tr>
<tr>
<td>IBARRA, A.</td>
<td>I3.2</td>
</tr>
<tr>
<td>IBARRA, Angel</td>
<td>O1A.1, P1.103, P2.160, P2.162, P2.163, P3.006, P3.151, P3.203</td>
</tr>
<tr>
<td>ICHARD, Mathieu</td>
<td>P2.126</td>
</tr>
<tr>
<td>ICHIKAWA, Masahiro</td>
<td>O2C.2</td>
</tr>
<tr>
<td>IDE, Shunsuke</td>
<td>P1.041</td>
</tr>
<tr>
<td>IDEI, Hiroshi</td>
<td>P1.046, P2.078</td>
</tr>
<tr>
<td>IGIELSKI, Andrzej</td>
<td>P2.053</td>
</tr>
<tr>
<td>IGITKHANOV, Y.</td>
<td>O5A.1</td>
</tr>
<tr>
<td>IGITKHANOV, Yuri</td>
<td>P1.113, P3.012</td>
</tr>
<tr>
<td>IGLESIAS, Daniel</td>
<td>P4.119</td>
</tr>
<tr>
<td>IGLESIAS, Silvia</td>
<td>P1.049, P3.056, P3.070</td>
</tr>
<tr>
<td>IGNACIO, Prieto Diaz</td>
<td>P3.202</td>
</tr>
<tr>
<td>IIHRA, Ryota</td>
<td>P1.180</td>
</tr>
<tr>
<td>IJIMA, Takaaki</td>
<td>P4.031, P4.112, P4.118</td>
</tr>
<tr>
<td>IKEDA, Katsunori</td>
<td>P4.032</td>
</tr>
<tr>
<td>IKEDA, Ryosuka</td>
<td>P3.023</td>
</tr>
<tr>
<td>IKEDA, Ryosuke</td>
<td>P1.028, P3.022</td>
</tr>
<tr>
<td>IKEDA, Y.</td>
<td>I3.3</td>
</tr>
<tr>
<td>ILHAN, Zeki</td>
<td>P3.042</td>
</tr>
<tr>
<td>ILKEI, Tamas</td>
<td>P3.053</td>
</tr>
<tr>
<td>ILLY, Stefan</td>
<td>O5B.3, P3.025</td>
</tr>
<tr>
<td>IM, Kihak</td>
<td>P1.202, P2.118, P2.119, P2.120, P2.159</td>
</tr>
<tr>
<td>IMAGAWA, Shinsaku</td>
<td>P1.004</td>
</tr>
<tr>
<td>IMRISEK, Martin</td>
<td>P2.047</td>
</tr>
<tr>
<td>IN, Sang-Ryul</td>
<td>P4.026</td>
</tr>
<tr>
<td>IN, Yong-kyoon</td>
<td>O1B.2</td>
</tr>
<tr>
<td>INCELLI, Marco</td>
<td>P4.148, P4.149, P4.150</td>
</tr>
</tbody>
</table>
INFANTE, Virginia  
INGESSON, L. C.  
INJUTIN, Nikolai  
INNOCENTE, Paolo  
INOMIYA, Dai  
INOUE, Daisuke  
INOUE, Ryuichi  
IOANNIDIS, Zisis  
IOANNIDIS, Zisis C.  
IONITA, Codrina  
ISAYAMA, Akihiko  
ISHIKAWA, Fumitaka  
ISHIKAWA, Humitaka  
ISOBAR, Mitsutaka  
ISOBE, Kanetsugu  
ISONO, Takaaki  
ISOZAKI, Masami  
ITAMI, Kiyoshi  
ITEN, Michael  
ITO, Satoshi  
ITURRIZA, Inigo  
IVANOV, A.  
IVANOV, Alexander A.  
IVANOVA-STANIK, Irena  
IVANTSIVSKY, Maksim  
IVEKOVIĆ, Alajaz  
IVEKOVIĆ, Aljaz  
IWAI, Yasunori  
IWAMA, Yuki  
IWAMOTO, Akifumi  
IWATA, Noriyuki Y.  
JABLONSKI, Slawomir  
JABOULAY, J-Ch.  
JACQUET, Philippe  
JACQUOT  
JADHAV, M.  
JADHAV, Mahesh  
JAESIC, Hong  
JAGANNATHAN, Govindarajan  
JAGER, Ales  
JAGER, Martin  
JAKSIC, Nikola  
JAKUBOWSKI, Marcin  
JANA, Mukti Ranjan  
JANA, Subrata  
JANG, Jae-young  
JANKY, Filip  
JANOSI, Laszlo  
JASPER, Bruno  
JAYASWAL, Snehal P.  
JEDNOROG, S.  
JELONNEK, John  
JENKINS, Ian  
JENUS, Petra  
JENZSCH, Hartmut  
JEON, Young-Mu  
JEONG, Jinhun  
JEONG, Nam-yong  
JEONG, Seong-Jo  
JEONG, Seung Ho  
JEONG, Seung-Ho
<table>
<thead>
<tr>
<th>Name</th>
<th>Page Numbers</th>
</tr>
</thead>
<tbody>
<tr>
<td>JEONG, Yong Hwang</td>
<td>P1.188</td>
</tr>
<tr>
<td>JET CONTRIBUTORS</td>
<td>I2.1, I5.5, P1.044, P1.045</td>
</tr>
<tr>
<td>JET, Contributors</td>
<td>P4.120</td>
</tr>
<tr>
<td>JET, contributors</td>
<td>P1.060, P1.064, P3.048, P4.036</td>
</tr>
<tr>
<td>JHA, Akhil</td>
<td>P4.034, P4.035</td>
</tr>
<tr>
<td>JHANG, Hogun</td>
<td>P4.040</td>
</tr>
<tr>
<td>JI, Xiang</td>
<td>O4C.1</td>
</tr>
<tr>
<td>JI, Xiaoquan</td>
<td>P2.138</td>
</tr>
<tr>
<td>JIA, Jiangtao</td>
<td>P2.203</td>
</tr>
<tr>
<td>JIA, Shenli</td>
<td>P2.095, P2.096</td>
</tr>
<tr>
<td>JIAN FEI, Peng</td>
<td>P4.094</td>
</tr>
<tr>
<td>JIANG, Baoqiang</td>
<td>P4.095</td>
</tr>
<tr>
<td>JIANG, Hualei</td>
<td>P2.202</td>
</tr>
<tr>
<td>JIANG, Jiamei</td>
<td>P2.138, P4.157</td>
</tr>
<tr>
<td>JIANG, Jiejiong</td>
<td>O4C.1, P1.199, P2.203</td>
</tr>
<tr>
<td>JIANG, Kecheng</td>
<td>P2.154, P2.156, P2.158</td>
</tr>
<tr>
<td>JIANG, Tao</td>
<td>P1.005, P4.157</td>
</tr>
<tr>
<td>JIANG, W.</td>
<td>P2.076</td>
</tr>
<tr>
<td>JIANG, Yanzheng</td>
<td>P1.046</td>
</tr>
<tr>
<td>JIANYONG, Cao</td>
<td>P4.030</td>
</tr>
<tr>
<td>JIEFENG, Wu</td>
<td>P4.140</td>
</tr>
<tr>
<td>JILEK, Richard</td>
<td>O4B.4, P2.112</td>
</tr>
<tr>
<td>JIMENEZ REY, David</td>
<td>P3.008</td>
</tr>
<tr>
<td>JIN, Cheng</td>
<td>P1.158, P1.159</td>
</tr>
<tr>
<td>JIN, Hyoung Gon</td>
<td>P1.166</td>
</tr>
<tr>
<td>JIN, Hyung Gon</td>
<td>P1.165, P3.150, P4.106</td>
</tr>
<tr>
<td>JIN, Jianbo</td>
<td>O5B.3, P3.025</td>
</tr>
<tr>
<td>JIN, Sung-Wook</td>
<td>P1.130</td>
</tr>
<tr>
<td>JIN, Xue Zhou</td>
<td>O4C.2, P2.198</td>
</tr>
<tr>
<td>JING, Song</td>
<td>P3.205</td>
</tr>
<tr>
<td>JINHONG, Yang</td>
<td>P1.068</td>
</tr>
<tr>
<td>JIOLAT, Guillaume</td>
<td>P2.091</td>
</tr>
<tr>
<td>JIRSA, Milos</td>
<td>P4.073</td>
</tr>
<tr>
<td>JODŁOWSKI, Paweł</td>
<td>P1.185</td>
</tr>
<tr>
<td>JOGI, Indrek</td>
<td>P4.066</td>
</tr>
<tr>
<td>JOHNSON, D.</td>
<td>P4.057</td>
</tr>
<tr>
<td>JOHNSON, Robert D.</td>
<td>O5C.4</td>
</tr>
<tr>
<td>JOHNSTON, Jane</td>
<td>P3.197</td>
</tr>
<tr>
<td>JOISA, Y. Sankar</td>
<td>P3.015</td>
</tr>
<tr>
<td>JONES, G.</td>
<td>P4.169</td>
</tr>
<tr>
<td>JONES, Graham</td>
<td>P2.003</td>
</tr>
<tr>
<td>JONES, Luke</td>
<td>P2.004</td>
</tr>
<tr>
<td>JONES, Timothy</td>
<td>P2.003</td>
</tr>
<tr>
<td>JONG, A.</td>
<td>P3.161</td>
</tr>
<tr>
<td>JOO, Jae-Joon</td>
<td>P4.002, P4.004</td>
</tr>
<tr>
<td>JOSHI, Hemant</td>
<td>P1.069, P1.070</td>
</tr>
<tr>
<td>JOSHI, Jaydeepkumar</td>
<td>P2.027</td>
</tr>
<tr>
<td>JOULIA, Xavier</td>
<td>P4.147</td>
</tr>
<tr>
<td>JOUVE, Michel</td>
<td>P1.069, P1.071</td>
</tr>
<tr>
<td>JT-60SA TEAM</td>
<td>I1.3</td>
</tr>
<tr>
<td>JUAREZ, Rafael</td>
<td>P3.203</td>
</tr>
<tr>
<td>JUHERA, Eduard</td>
<td>P4.165, P4.192</td>
</tr>
<tr>
<td>JUN, Tao</td>
<td>P4.046</td>
</tr>
<tr>
<td>JUNG, Bong-Ki</td>
<td>P4.027</td>
</tr>
<tr>
<td>JUNG, Bongki</td>
<td>P1.080, P4.026</td>
</tr>
<tr>
<td>JUNG, Hee Joon</td>
<td>O3B.4</td>
</tr>
<tr>
<td>JUNG, Hun-Cheol</td>
<td>P4.125</td>
</tr>
<tr>
<td>JUNG, Hunchea</td>
<td>P2.143</td>
</tr>
<tr>
<td>JUNG, Ki-Jung</td>
<td>P1.127</td>
</tr>
<tr>
<td>JUNG, Kwangjin</td>
<td>P4.153, P4.154</td>
</tr>
<tr>
<td>JUNG, Laurent</td>
<td>P4.040</td>
</tr>
<tr>
<td>JUNGHANNS, Patrick</td>
<td>P2.124, P2.125</td>
</tr>
<tr>
<td>JUNLI, Qi</td>
<td>P1.068</td>
</tr>
<tr>
<td>KACZMARCZYK, Jacek</td>
<td>P2.066, P2.069, P2.070</td>
</tr>
</tbody>
</table>
KAGINAKA, Masaru P4.177
KAIFU, Gan P1.068
KAISER, Benedikt P3.116
KAJI, Sayaka P4.181
KAJITANI, Hideki P3.078
KAKUDATE, Satoshi P4.128, P4.129, P4.133
KALARIA, Parth O5B.3
KALIATKA, Tadas P1.204
KALININ, Vladimir P4.012
KALLENBACH, Arne O5C.3, P1.038, P1.111
KALLMEYER, Johannes Peter P1.013
KALLMEYER, Peter P2.063
KALSEY, Manninder P4.020
KAMADA, Y. I1.3
KAMIYA, K. P1.074
KAMLAH, Marc P1.157, P3.171
KAMPF, Dirk P1.058, P4.055, P4.056
KANDA, Kazuhiro P1.020
KANDAUROV, Igor P3.125
KANEKO, Osamu P4.032
KANEMURA, Takuji P4.173, P4.175, P4.177, P4.181
KANG, Bo Ram P3.106
KANG, Dong Kwon P1.127, P1.132
KANG, Hee-Seok P4.153
KANG, Hee-Suk P4.154
KANG, Kyoung-O P1.127, P1.132
KANG, Qinlang P1.152
KANG, Suk-Hoon P1.188
KANG, Youngkil P1.127
KANNAMULLER, Mario O3A.2
KANOONGO, Nitin P2.026
KAPARKOVA, Marina P4.011, P4.012
KAPITONOV, V. P4.038
KAPRANOV, Ilya P3.088
KARASKOVA NENADALOVA, Lucie P1.205
KARPOV, Aleksey P3.124
KARPUSHOV, Alexander P3.027
KARPUSHOV, Alexander N. P3.028, P3.030
KASADA, Ryuta P1.149, P1.180, P1.182, P1.207, P2.166, P3.121, P3.174
KASATOV, Alexander P3.125
KASEMANN, Claus-Peter P1.038
KASHIWAGI, Mieko O2C.2, P2.028
KASPARSEK, Walter P3.025, P4.124
KASPROWICZ, Grzegorz P3.073, P4.068, P4.070, P4.071
KASTHIRIRENGAN, S. P4.163
KATAYAMA, Kazunari P1.206, P3.167, P4.115
KATO, Taichiro P1.181
KAUFMAN, Michael P4.036, P4.037
KAVIN, Andrey P3.086
KAWABATA, Yoshiya P1.206
KAWAMATA, Yoichi P1.046, P1.073
KAWAMURA, Kazutaka P4.031, P4.112, P4.118
KAWAMURA, Yoshinori P3.164
KAWASAKI, Shoji P1.046, P2.078
KEENAN, T. P4.169
KEENAN, Tom P2.003, P4.170
KEEP, Jonathan O3C.3, P3.140
KELLER, Dephine P4.137
KELLMAN, A. P2.035
KELLMAN, D. P2.035
KEMBLETON, Richard P1.082
KEMBLETON, Richard P2.192, P4.020
KEMP, Richard O1B.1, P2.194, P4.003, P4.003
KENNEDY, Cameron O3C.1
Khan, Mohammad Shoaib  
P3.015
Khan, Ziauddin  
P3.015
Khayrutdinov, Rustam  
P2.045
Khirwadkar, Samir  
P4.108
Khodak, Andrei  
P2.154, P2.167, P3.054, P3.158, P3.201
Kokhlovin, Mikhail  
P2.104
Komikov, Sergey  
P2.130
Khiripunov, Boris  
P3.125
Khiripunov, Vladimir  
P3.104, P4.016
Khristi, Yohan S.  
P3.015
Kivostenko, Peter  
P2.045
Kivostenko, Petr  
P1.094
Kidambi, Rajamannar Swamy  
P4.108
Kikuchi, Akihiro  
P1.081
Kilpelainen, Pekka  
P3.052
Kim, Do Hyun  
P1.202
Kim, Dong Jun  
P2.135, P4.105
Kim, Geon-Woo  
P2.159
Kim, Gwang-Ho  
P1.130
Kim, H. T.  
P4.090
Kim, H.C.  
P4.104
Kim, H.T.  
P4.104
Kim, Haejin  
P2.030, P2.031, P2.032
Kim, Hak-Kun  
P1.130
Kim, Hee-Soo  
P2.074
Kim, Heung-Su  
P2.046
Kim, Hong Tack  
P2.117
Kim, Hong-Tack  
P4.001, P4.006, P4.159
Kim, Hyoung Chan  
O2A.2, P1.190, P1.191
Kim, Hyun-Soo  
P1.130
Kim, Hyungdae  
P4.103
Kim, Jae Hyun  
P1.202
Kim, Jae-Hwan  
P1.161
Kim, Jeenhyun  
P2.032, P2.033
Kim, Jong Su  
P4.005
Kim, Jongsu  
P4.001
Kim, Kwang-Pyo  
P4.001
Kim, Kyung-Min  
P4.104
Kim, Kyungmin  
P2.117, P4.006
Kim, Nam-Won  
P4.002, P4.004
Kim, S.K.  
P4.104
Kim, Sa-Woong  
P4.125
Kim, Sang-Tae  
O1B.2
Kim, Sawoong  
P2.143
Kim, Seongcheol  
P2.043
Kim, Song Hyun  
P1.202
Kim, Suk-Kwon  
P1.166, P3.150, P4.106
Kim, Sun-Ho  
P4.027
Kim, Tae-Seok  
P1.130
Kim, Tae-Seong  
P4.005, P4.026, P4.027
Kim, Taejoon  
P3.183
Kim, Y. H.  
P4.090
Kim, Y. O.  
P4.090
Kim, Yaeung-Soo  
O1B.2, P4.002, P4.004
Kim, Yaungsoo  
P4.001
Kim, Yeannin  
P4.153, P4.154
Kim, Yoo-Sung  
P1.077, P1.078, P1.080
Kim, You Bean  
P4.205
Kim, Young-Gi  
P1.079
Kim, Yu-Gyeong  
P2.134
Kimura, Akihiko  
P1.182
KING, B. Damian  
O5A.2
KINNA, David  
O4A.1
Kiptilky, V.  
P3.047, P3.048
<table>
<thead>
<tr>
<th>Name</th>
<th>Pages</th>
</tr>
</thead>
<tbody>
<tr>
<td>KROM, Jon G.</td>
<td>P3.037</td>
</tr>
<tr>
<td>KRUEZI, Uron</td>
<td>O4A.1, P4.121</td>
</tr>
<tr>
<td>KRUPIN, Vadim</td>
<td>P3.124</td>
</tr>
<tr>
<td>KRYCHOWIAK, Maciej</td>
<td>P2.065</td>
</tr>
<tr>
<td>KSIAZEK, Ireneusz</td>
<td>P2.066</td>
</tr>
<tr>
<td>KU, Duck Young</td>
<td>P1.192</td>
</tr>
<tr>
<td>KUBAK, Jindrich</td>
<td>P1.014</td>
</tr>
<tr>
<td>KUBKOWSKA, Monika</td>
<td>I5.1, P2.066, P2.069, P2.070</td>
</tr>
<tr>
<td>KULDACEK, Ondrej</td>
<td>P2.038</td>
</tr>
<tr>
<td>KUHNER, Georg</td>
<td>O5C.2, P2.062, P3.036, P3.037, P3.038</td>
</tr>
<tr>
<td>KUKHTIN, Vladimir</td>
<td>P4.010, P4.010, P4.011</td>
</tr>
<tr>
<td>KUKUSHIKIN, Alexander B.</td>
<td>P4.053</td>
</tr>
<tr>
<td>KUKUSHIKIN, Andrei</td>
<td>P4.050</td>
</tr>
<tr>
<td>KULSARTOV, Timur</td>
<td>P1.109, P4.116</td>
</tr>
<tr>
<td>KUMAKI, Takuya</td>
<td>P1.004</td>
</tr>
<tr>
<td>KUMAR, A. G. Ajith</td>
<td>P1.001</td>
</tr>
<tr>
<td>KUMAR, Ajith</td>
<td>P1.006</td>
</tr>
<tr>
<td>KUMAR, Aveg</td>
<td>P1.070</td>
</tr>
<tr>
<td>KUMAR, Manoj</td>
<td>P3.015</td>
</tr>
<tr>
<td>KUMAR, Mohit</td>
<td>P1.006</td>
</tr>
<tr>
<td>KUMAR, Punit</td>
<td>P2.207</td>
</tr>
<tr>
<td>KUMAR, Rajnish</td>
<td>P4.034, P4.035</td>
</tr>
<tr>
<td>KUMAR, Ratnesh</td>
<td>P3.002</td>
</tr>
<tr>
<td>KUMAR, Varunesh</td>
<td>P4.013</td>
</tr>
<tr>
<td>KUMARI, Praveena</td>
<td>P1.070</td>
</tr>
<tr>
<td>KUNG, Chun</td>
<td>P3.035</td>
</tr>
<tr>
<td>KUNO, Kazuo</td>
<td>P2.085</td>
</tr>
<tr>
<td>KUNZE, Andre</td>
<td>P2.113</td>
</tr>
<tr>
<td>KUPRIYANOV, Igor</td>
<td>P2.173</td>
</tr>
<tr>
<td>KURATA, Rie</td>
<td>P4.143</td>
</tr>
<tr>
<td>KURIHARA, Kenichi</td>
<td>P1.046, P1.073</td>
</tr>
<tr>
<td>KUROTAKI, Hironori</td>
<td>P1.178, P4.120</td>
</tr>
<tr>
<td>KUROWSKI, Arkadiusz</td>
<td>P2.053</td>
</tr>
<tr>
<td>KUROYANAGI, Shinpei</td>
<td>P3.102</td>
</tr>
<tr>
<td>KURSKIEV, Gleb</td>
<td>P4.058</td>
</tr>
<tr>
<td>KURTZ, Richard J</td>
<td>O3B.4</td>
</tr>
<tr>
<td>KUTEEV, Boris</td>
<td>P1.150</td>
</tr>
<tr>
<td>KUZMINA, Galina</td>
<td>P2.045</td>
</tr>
<tr>
<td>KUZOVKOV, Vladimir</td>
<td>P3.184</td>
</tr>
<tr>
<td>KWAG, Sang-Woo</td>
<td>P4.002, P4.004</td>
</tr>
<tr>
<td>KWAK, Jong gu</td>
<td>P2.031</td>
</tr>
<tr>
<td>KWAK, Jong-Gu</td>
<td>O1B.2, P2.029, P2.074</td>
</tr>
<tr>
<td>KWIAKTOWSKI, Roch</td>
<td>P3.045, P3.047</td>
</tr>
<tr>
<td>KWNA, Sang Woo</td>
<td>P4.001</td>
</tr>
<tr>
<td>KWON, Gil</td>
<td>P3.202</td>
</tr>
<tr>
<td>KWON, Saerom</td>
<td>P3.176, P3.178</td>
</tr>
<tr>
<td>KWON, Sungjin</td>
<td>P1.202, P2.118, P2.119, P2.120</td>
</tr>
<tr>
<td>KYLE-HENNEY, Stephen</td>
<td>P2.174</td>
</tr>
<tr>
<td>KYRIELEIS, Albrecht</td>
<td>P1.129</td>
</tr>
<tr>
<td>KLOSOWSKI, Mariusz</td>
<td>P2.004</td>
</tr>
<tr>
<td>LAAD, Rahul</td>
<td>P1.184</td>
</tr>
<tr>
<td>LAAN, Matti</td>
<td>P4.066</td>
</tr>
<tr>
<td>LABASSE, Florence</td>
<td>O3A.4</td>
</tr>
<tr>
<td>LABATE, Carmelenzo</td>
<td>P1.023</td>
</tr>
<tr>
<td>LACES, Selece</td>
<td>P1.103</td>
</tr>
<tr>
<td>LACKNER, Karl</td>
<td>O5C.3</td>
</tr>
<tr>
<td>LACROIX, Benoit</td>
<td>P1.088, P2.087</td>
</tr>
<tr>
<td>LAGOS, Pedro</td>
<td>O3A.1, P1.053, P2.049</td>
</tr>
<tr>
<td>LAGUTKINA, Anna</td>
<td>P2.136</td>
</tr>
<tr>
<td>LAM, N.</td>
<td>P3.048</td>
</tr>
<tr>
<td>LAM, Norman</td>
<td>P3.051</td>
</tr>
<tr>
<td>LAMALLE, Philippe</td>
<td>P2.131</td>
</tr>
<tr>
<td>LAMBERTZ, Horst</td>
<td>O4A.1</td>
</tr>
<tr>
<td>LAMIKIZ, Aitzol</td>
<td>P2.128</td>
</tr>
</tbody>
</table>
LAMPASI, Alessandro  P2.011, P2.081, P2.082
LAMZIN, Evgeny   P4.010, P4.010, P4.011
LAN, T.          P2.076
LANCETOV, Andrey P2.171
LANCHI, Claudia  P2.010, P2.010
LANDIS, Jean-Daniel P3.019, P3.020, P3.021
LANDMAN, Igor    P3.199
LANG, P. T.      O5A.1
LANG, Peter      P2.019, P4.047
LANG, Peter T.   P4.161
LANG, Peter Thomas P3.012
LANGE, Christian P1.096
LANGESLAG, Stefanie Agnes Elisabeth P1.187
LANGHORN, A.     P3.081
LAQUA, Heike      P2.062, P3.036, P3.037
LAQUA, Heique    P1.070
LARROQUE, Sebastien O3A.4, P1.136, P3.123
LASHUKOV, Alexander P3.086
LASNIER, C.       O4B.3
LASZYNSKA, E      P1.196
LASZYNSKA, E.     P1.066
LATERZA, Bruno   P4.007
LATASAS, George P. O5B.3
LATASAS, Georges P3.025
LAURET, Menno    P3.041
LAVANCHY, Pierre P3.030
LAWLESS, R.      O5A.1
LAZEA-STOYANOVA, Andrada P3.108
LAZERSON, Samuel  P2.094
LAZZARO, Gabriele P1.025
LE COZ, Quentin   P1.088
LE FOL, Frederic P4.054
LE GUERN, Frederic P3.054, P4.054
LECHTE, Carsten   P3.025
LEDDA, Francesco P1.044, P2.039, P4.044
LEE, Chang-Hoon  O2A.2, P1.190, P1.191
LEE, Chul-Hee    P4.002, P4.004
LEE, D.W.        P4.104
LEE, Dong Won    P1.165, P1.166, P2.135, P3.150, P4.105, P4.106
LEE, Dongwon     P1.188
LEE, Eo Hwak     P1.166, P3.150, P4.106
LEE, Gisik       P3.202
LEE, H.Y.        P1.079
LEE, Heun Tae    P1.110, P3.101, P3.153
LEE, Hyeon Gon   P1.127, P4.059
LEE, Hyeon-Gon   P1.130
LEE, Hyeun Gon   P2.127
LEE, J.          P1.079
LEE, J.H.        P1.079
LEE, Jeong-Hun   P2.159
LEE, Jeongwon    P2.043
LEE, Jin-Jong    P1.190
LEE, Jong-ha     P2.075
LEE, Jong-hwa    P4.006, P4.159
LEE, Juhyoun     P4.006, P4.159
LEE, Kwang Won   P4.005, P4.026
LEE, Si-Woo      P1.166
LEE, Suong-hun   P2.075
LEE, Tae-Ho     O2A.2, P1.190, P1.191
LEE, Taegu      P3.202
LEE, William E.  P3.107
LEE, Woongryol   P3.202
<table>
<thead>
<tr>
<th>Name</th>
<th>Codes</th>
</tr>
</thead>
<tbody>
<tr>
<td>LEE, Young-Ju</td>
<td>P4.002, P4.004</td>
</tr>
<tr>
<td>LEE, Youngmin</td>
<td>P1.165, P1.189, P1.192, P3.165</td>
</tr>
<tr>
<td>LEE, Youngseok</td>
<td>P2.074</td>
</tr>
<tr>
<td>LEFEVBRE, X.</td>
<td>I5.5</td>
</tr>
<tr>
<td>LEGARDA, Fernando</td>
<td>P4.191</td>
</tr>
<tr>
<td>LEGUERN, Frederic</td>
<td>O3A.4</td>
</tr>
<tr>
<td>LEGUEY, Teresa</td>
<td>P1.174</td>
</tr>
<tr>
<td>LEHNEN, Michael</td>
<td>P4.046</td>
</tr>
<tr>
<td>LEI, Lei</td>
<td>P4.082</td>
</tr>
<tr>
<td>LEI, Mingzhu</td>
<td>P1.121, P1.162</td>
</tr>
<tr>
<td>LEICHTLE, D.</td>
<td>I3.3, P1.196</td>
</tr>
<tr>
<td>LEICHTLE, Dieter</td>
<td>P2.131</td>
</tr>
<tr>
<td>LEIPOLD, Frank</td>
<td>P3.051</td>
</tr>
<tr>
<td>LEITENSTERN, Peter</td>
<td>P4.083</td>
</tr>
<tr>
<td>LENGAR, I.</td>
<td>P1.066, P3.047, P3.048</td>
</tr>
<tr>
<td>LENGAR, Igor</td>
<td>P3.046</td>
</tr>
<tr>
<td>LENNHOLM, Morten</td>
<td>P1.045, P4.047</td>
</tr>
<tr>
<td>LEONARD, A.W.</td>
<td>O4B.3</td>
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<tr>
<td>LEONHARDT, Wolfgang</td>
<td>P1.026</td>
</tr>
<tr>
<td>LEONID, Zakharov</td>
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</tr>
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<td>LERCHE, Ernesto</td>
<td>P1.045, P4.036, P4.037</td>
</tr>
<tr>
<td>LESHUKOV, Andrey</td>
<td>P2.104, P2.106</td>
</tr>
<tr>
<td>LESSARD, Timothy</td>
<td>P4.113</td>
</tr>
<tr>
<td>LETELLIER, Laurent</td>
<td>O3A.4</td>
</tr>
<tr>
<td>LEVASHOVA, Maria</td>
<td>P4.050</td>
</tr>
<tr>
<td>LEWANDOWSKA, Monika</td>
<td>P1.083, P1.087, P2.196</td>
</tr>
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<td>LEWENTZ, Marc</td>
<td>P1.070, P2.062, P3.036, P3.037</td>
</tr>
<tr>
<td>LEWS, Oliver</td>
<td>P3.170, P4.171</td>
</tr>
<tr>
<td>LI PUMA, Antonella</td>
<td>O5A.4</td>
</tr>
<tr>
<td>LI, Bo</td>
<td>P1.005, P1.008, P4.157</td>
</tr>
<tr>
<td>LI, Changzhen</td>
<td>P4.159</td>
</tr>
<tr>
<td>LI, Chuan</td>
<td>P3.076</td>
</tr>
<tr>
<td>LI, Guoqiang</td>
<td>O2C.4</td>
</tr>
<tr>
<td>LI, Huajun</td>
<td>P4.095</td>
</tr>
<tr>
<td>LI, Jia</td>
<td>P1.160, P3.204</td>
</tr>
<tr>
<td>LI, Jianguang</td>
<td>I2.2, O2B.4, O2C.4</td>
</tr>
<tr>
<td>LI, Jiaxian</td>
<td>P4.049</td>
</tr>
<tr>
<td>LI, Kang</td>
<td>P4.094</td>
</tr>
<tr>
<td>LI, Ming</td>
<td>P1.143, P1.144, P3.138, P4.127</td>
</tr>
<tr>
<td>LI, Muyuan</td>
<td>P1.118, P2.114</td>
</tr>
<tr>
<td>LI, Pan</td>
<td>P2.157</td>
</tr>
<tr>
<td>LI, Q.</td>
<td>P4.101</td>
</tr>
<tr>
<td>LI, Qian</td>
<td>P2.187</td>
</tr>
<tr>
<td>LI, Sheng</td>
<td>P2.095, P2.096</td>
</tr>
<tr>
<td>LI, Taosheng</td>
<td>O4C.1</td>
</tr>
<tr>
<td>LI, Wei</td>
<td>P1.005, P1.008, P3.163, P4.084, P4.157</td>
</tr>
<tr>
<td>LI, Yuanjie</td>
<td>P1.160</td>
</tr>
<tr>
<td>LI, Zaixin</td>
<td>P4.201</td>
</tr>
<tr>
<td>LIANG, L.</td>
<td>P2.036</td>
</tr>
<tr>
<td>LIAO, Yuanxu</td>
<td>P3.076</td>
</tr>
<tr>
<td>LIBEYRE, Paul</td>
<td>P1.187</td>
</tr>
<tr>
<td>LIENIG, Tim</td>
<td>P3.192</td>
</tr>
<tr>
<td>LIEVIN, Christophe</td>
<td>P4.008</td>
</tr>
<tr>
<td>LIGER, Karine</td>
<td>O4C.2, P2.017, P4.147</td>
</tr>
<tr>
<td>LIKONEN, Jari</td>
<td>I2.1</td>
</tr>
<tr>
<td>LIKONEN, Jari</td>
<td>P1.108</td>
</tr>
<tr>
<td>LILLLEY, S.</td>
<td>P1.196</td>
</tr>
<tr>
<td>LIM, Jongmin</td>
<td>P1.132</td>
</tr>
<tr>
<td>LIM, Kisuk</td>
<td>P1.127</td>
</tr>
<tr>
<td>LIN, Tao</td>
<td>P1.122</td>
</tr>
<tr>
<td>LINCZUK, Pawel</td>
<td>P3.073</td>
</tr>
<tr>
<td>LINIERS, Macarena</td>
<td>P4.029</td>
</tr>
<tr>
<td>LINKE, Jochen</td>
<td>O5B.1</td>
</tr>
</tbody>
</table>
LU, Kun P3.195
LU, L. I3.3
LU, Lei P1.129, P1.153, P3.200
LU, Mingxuan P1.121
LU, Yong P1.122
LUCA, Grando P1.024
LUCE, Timothy C. O5C.4
LUCHETTA, Adriano P1.021, P1.023, P1.025, P4.009
LUCHETTA, Adriano Francesco P3.069
LUCULESCU, Catalin P3.111
LUDWIG, Alfred P2.180
LUIS, Raul O3A.1, P1.053, P2.049
LUKAC, Frantisek O2A.3
LUKASH, Victor P2.045
LUKIN, Alexander P4.006, P4.159
LUKIN, Alexey P2.132
LUKYYANOV, Vitaly P2.136
LUMSDAINE, Arnold P2.122, P4.113
LUNGU, Cristian P4.066
LUNGU, Cristian P. P3.110
LUNGU, Cristian Petrica P3.111
LUNGU, Mihail P3.111
LUNT, Tilmann O5C.3
Luo, Deli P3.162
Luo, G.-N. P4.101
Luo, Wenhua P3.162
Luo, X. O5A.1
Luo, Zhengping P2.042
Lupelli, Ivan P2.042, P4.062, P4.076
Luptakova, Natalia P3.191
Lushchik, Aleksandr P2.190, P3.184
Lux, Hanni O1B.1, P2.192
Lv, Zhongliang P3.163, P4.084
Lyssioivan, Anatoli P3.034
Lyu, B. P1.067
Lyublin, Boris P4.011
Lyytinen, Janne P3.052
Ma, Jianguo P4.141
Ma, Rui P4.049
Ma, Xuebin P2.154, P2.156, P2.158
Maatta, Timo P3.052
Mackel, Felix P4.077
Maddaluno, Giorgio P2.008
Maebara, Sunao P2.209
Maeji, Takeru P3.102, P3.103
Maejima, Tetsuya P2.028
Maffia, Giuseppe P2.011
Magaud, Philippe P1.118, P1.137
Maggio, Daniele P4.204
Maggiora, Edoardo P1.022
Maggiora, R. I4.3
Maggiora, Riccardo P2.007
Magielsen, Lida P1.146, P3.161
Mahajan, Kirti P3.015
Maheshwari, Abha P1.186
Mahesuriya, Gaurang I. P3.015
Maier, Hans O4B.1, P4.123
Maione, Ivan P3.127
Maione, Ivan A. P3.120
Maione, Ivan Alessio P3.122
Maj, Adam P3.006
Makarov, Oleg P1.173
Makhankov, Aleksey P3.124
Makinen, Harri P3.052
<table>
<thead>
<tr>
<th>Name</th>
<th>Paper Number(s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>MAKWANA, Azadsinh R.</td>
<td>P3.015</td>
</tr>
<tr>
<td>MALAGIUS, Artur</td>
<td>P4.065</td>
</tr>
<tr>
<td>MALAVASI, Andrea</td>
<td>P2.012, P2.014</td>
</tr>
<tr>
<td>MALEC, Stanislav</td>
<td>P2.109</td>
</tr>
<tr>
<td>MALINOWSKI, Karol</td>
<td>P4.068, P4.070, P4.071</td>
</tr>
<tr>
<td>MALINOWSKI, Leszek</td>
<td>P1.083, P2.196</td>
</tr>
<tr>
<td>MALIZIA, Andrea</td>
<td>P1.204, P4.196, P4.197, P4.198</td>
</tr>
<tr>
<td>MALJAARS, Bert</td>
<td>P2.044, P4.047</td>
</tr>
<tr>
<td>MALO, Marta</td>
<td>O3B.3, P1.154, P1.155, P3.146, P3.172, P3.175</td>
</tr>
<tr>
<td>MALOUCH, F.</td>
<td>I3.3</td>
</tr>
<tr>
<td>MALOUCH, Fadhel</td>
<td>O5A.3</td>
</tr>
<tr>
<td>MAMCHITS, Dmitry</td>
<td>P2.136</td>
</tr>
<tr>
<td>MANCHANDA, Ranjana</td>
<td>P3.015</td>
</tr>
<tr>
<td>MANCINI, A.</td>
<td>I4.3</td>
</tr>
<tr>
<td>MANCINI, Andrea</td>
<td>P2.051</td>
</tr>
<tr>
<td>MANCUSI, Davide</td>
<td>O5A.3</td>
</tr>
<tr>
<td>MANDAR, Hugo</td>
<td>P2.190</td>
</tr>
<tr>
<td>MANDUCHI, Gabriele</td>
<td>P2.038, P3.069</td>
</tr>
<tr>
<td>MANIERO, Moreno</td>
<td>P1.025</td>
</tr>
<tr>
<td>MANK, Klaus</td>
<td>P4.080</td>
</tr>
<tr>
<td>MANSURI, Imran</td>
<td>P1.070</td>
</tr>
<tr>
<td>MANSURI, Imran A.</td>
<td>P3.015</td>
</tr>
<tr>
<td>MANTEL, Nicolas</td>
<td>P4.020</td>
</tr>
<tr>
<td>MANZANARES, Ana</td>
<td>O3A.4</td>
</tr>
<tr>
<td>MANZUK, Maksim</td>
<td>P3.087</td>
</tr>
<tr>
<td>MAO, Bingyan</td>
<td>P1.144, P4.127, P4.134</td>
</tr>
<tr>
<td>MAO, Shifeng</td>
<td>P1.015, P1.016, P1.162</td>
</tr>
<tr>
<td>MAO, Weicheng</td>
<td>P4.095</td>
</tr>
<tr>
<td>MAO, Xin</td>
<td>P1.123</td>
</tr>
<tr>
<td>MAO, Yiran</td>
<td>O2A.4</td>
</tr>
<tr>
<td>MAQUEDA, Luis</td>
<td>P2.160, P2.161, P2.163</td>
</tr>
<tr>
<td>MAQUET, Philippe</td>
<td>P3.057, P4.054, P4.124</td>
</tr>
<tr>
<td>MARASCHEK, Marc</td>
<td>P4.047</td>
</tr>
<tr>
<td>MARASCU, Valentina</td>
<td>P3.108</td>
</tr>
<tr>
<td>MARCHIORI, Giuseppe</td>
<td>O1B.3, P1.075, P2.038, P2.039, P2.040, P2.041</td>
</tr>
<tr>
<td>MARCINKEVICIUS, Benjaminas</td>
<td>P2.053</td>
</tr>
<tr>
<td>MARCO, Tardocchi</td>
<td>P2.080</td>
</tr>
<tr>
<td>MARCONATO, Nicolo</td>
<td>P2.020, P2.021</td>
</tr>
<tr>
<td>MARCU, Aurelian</td>
<td>P3.111</td>
</tr>
<tr>
<td>MARCUZZI, Diego</td>
<td>P4.008</td>
</tr>
<tr>
<td>MARECHAL, Jean Louis</td>
<td>P2.091</td>
</tr>
<tr>
<td>MARIA TERESA, Porfiri</td>
<td>P4.195</td>
</tr>
<tr>
<td>MARIANO, Giovanni</td>
<td>O1A.4, P3.131</td>
</tr>
<tr>
<td>MARKOVIC, Tomas</td>
<td>P2.001, P2.047</td>
</tr>
<tr>
<td>MARLETAZ, Blaise</td>
<td>P3.028, P3.030</td>
</tr>
<tr>
<td>MARMILLOD, Philippe</td>
<td>P3.028, P3.030</td>
</tr>
<tr>
<td>MAROCCO, D.</td>
<td>P2.050</td>
</tr>
<tr>
<td>MAROCCO, Daniele</td>
<td>P2.051, P3.066</td>
</tr>
<tr>
<td>MARQUARDT, Mirko</td>
<td>P2.071</td>
</tr>
<tr>
<td>MARRELLI, Lionello</td>
<td>O1B.3, P2.038, P2.040</td>
</tr>
<tr>
<td>MARREN, C.</td>
<td>P3.048</td>
</tr>
<tr>
<td>MARSHALL, Roy</td>
<td>P2.002</td>
</tr>
<tr>
<td>MARTELLI, Daniele</td>
<td>P2.013, P4.190</td>
</tr>
<tr>
<td>MARTELLI, Emanuela</td>
<td>O1A.4, P2.148, P2.149, P3.135</td>
</tr>
<tr>
<td>MARTIKAINEN, Hannu</td>
<td>P3.052</td>
</tr>
<tr>
<td>MARTIN, Alex</td>
<td>P1.125, P1.129, P2.126</td>
</tr>
<tr>
<td>MARTIN, Fernando</td>
<td>P4.029</td>
</tr>
<tr>
<td>MARTIN, Vincent</td>
<td>O3A.4, P4.054</td>
</tr>
<tr>
<td>MARTIN, Yves</td>
<td>P3.030</td>
</tr>
<tr>
<td>MARTINEZ, Andre</td>
<td>P1.136</td>
</tr>
<tr>
<td>MARTINEZ, Emilii</td>
<td>P2.126</td>
</tr>
<tr>
<td>MARTINEZ, Jean Marc</td>
<td>P2.126</td>
</tr>
<tr>
<td>MARTINEZ-ESNAOLA, Jose Manuel</td>
<td>O3B.3</td>
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<td>P3.075</td>
</tr>
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<td>Code</td>
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<td>P1.044, P2.010, P2.039, P4.044</td>
</tr>
<tr>
<td>MARTOVETSKY, Nicolai</td>
<td>O1C.1, P3.080</td>
</tr>
<tr>
<td>MARTYENKO, Yuriy</td>
<td>P3.124</td>
</tr>
<tr>
<td>MARUYAMA, So</td>
<td>P1.005, P4.046, P4.157</td>
</tr>
<tr>
<td>MARUYAMA, Takahito</td>
<td>P4.128, P4.129, P4.133</td>
</tr>
<tr>
<td>MARZULLO, Domenico</td>
<td>P2.009, P2.141, P3.131, P3.132</td>
</tr>
<tr>
<td>MAS DE LES VALLS, Elisabet</td>
<td>P2.164</td>
</tr>
<tr>
<td>MAS SANCHEZ, Avelino</td>
<td>P3.019, P3.020, P3.021</td>
</tr>
<tr>
<td>MASAKI, Kei</td>
<td>O1B.4, P2.018</td>
</tr>
<tr>
<td>MASAND, Harish</td>
<td>P1.070, P3.015</td>
</tr>
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<td>MASASHI, Shimada</td>
<td>P3.114</td>
</tr>
<tr>
<td>MASCARADE, Jeremy</td>
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</tr>
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<td>MASIHELLO, Antonio</td>
<td>P4.008</td>
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<td>MASON, Mick</td>
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<td>P4.120</td>
</tr>
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<td>MATEJICEK, Jiri</td>
<td>O2A.3, P3.126, P3.192</td>
</tr>
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<td>P3.110</td>
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<td>MATSUDA, Shotaro</td>
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<td>MATSUKAWA, Makoto</td>
<td>P2.081, P2.082, P2.083</td>
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<td>MATSUMURA, Yoshihiro</td>
<td>P4.118</td>
</tr>
<tr>
<td>MATSUNAGA, Go</td>
<td>P1.073</td>
</tr>
<tr>
<td>MATSUURA, Hiroto</td>
<td>P3.173</td>
</tr>
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<td>MATSUYAMA, Akinobu</td>
<td>P1.041, P1.042</td>
</tr>
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<td>MATTEI, Massimiliano</td>
<td>P4.019, P4.044</td>
</tr>
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<td>O4A.1, P3.113, P4.061, P4.119</td>
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</tr>
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<td>P1.140</td>
</tr>
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<td>P4.068</td>
</tr>
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<td>MAZUL, Igor</td>
<td>P2.104, P3.124</td>
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<td>O4C.2, P1.204</td>
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<td>MAZZITELLI, Giuseppe</td>
<td>O3A.3, P1.109, P2.010</td>
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<td>MAZZONE, Giuseppe</td>
<td>P1.119, P1.120, P2.053, P2.141, P3.131, P3.132</td>
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<td>MCADAMS, Roy</td>
<td>P2.002, P2.003</td>
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<td>MCFARLAND, Adrian</td>
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<td>MCGINNIS, Dean</td>
<td>P2.122, P4.113</td>
</tr>
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<td>MCINTOSH, S.</td>
<td>P3.077</td>
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<td>MCINTOSH, Simon</td>
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<td>MCLEAN, A.</td>
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<td>MEDLEY, S. A.</td>
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<td>MEHTA, K.</td>
<td>O1C.2</td>
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<td>MEIER, Andreas</td>
<td>P1.054, P3.026</td>
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<td>MEISL, Gerd</td>
<td>P4.121</td>
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<td>MEISTER, Hans</td>
<td>O3A.2, P1.056, P1.057, P4.077, P4.081</td>
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<td>MEITNER, Steve</td>
<td>P4.015, P4.113</td>
</tr>
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<td>MELICHAR, Tomas</td>
<td>P2.160, P2.163, P3.119</td>
</tr>
<tr>
<td>MELISEK, Tibor</td>
<td>P4.073</td>
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<td>MELLEIN, Daniel</td>
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<td>MENARD, Jonathan E.</td>
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<td>MENEDEVITCH, Boris</td>
<td>P2.123, P2.124, P2.125</td>
</tr>
<tr>
<td>MENESSES, Luis</td>
<td>P4.086</td>
</tr>
<tr>
<td>MENG, Zi</td>
<td>P3.009</td>
</tr>
<tr>
<td>MENON, V.</td>
<td>P1.092</td>
</tr>
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<td>MERCADIER, Laurent</td>
<td>O4A.1</td>
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<td>P3.030</td>
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<td>Name</td>
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<td>MERRILL, Brad</td>
<td>O4C.3, P1.151</td>
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<td>P4.151</td>
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<td>MERTENS, Philippe</td>
<td>O4A.1, O4A.4, P4.060, P4.061, P4.078</td>
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<td>MERTENS, Vitus</td>
<td>P4.080</td>
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<tr>
<td>MESSIAEN, A.</td>
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<td>MESSIAEN, Andre</td>
<td>O5B.4, P1.034, P3.034</td>
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<td>MESTRE, R. Daniel</td>
<td>P4.137</td>
</tr>
<tr>
<td>MESZAROS, Botond</td>
<td>P2.140</td>
</tr>
<tr>
<td>METZ, Aron</td>
<td>P2.027</td>
</tr>
<tr>
<td>MEYER, Olivier</td>
<td>P1.070</td>
</tr>
<tr>
<td>MEYER, U.</td>
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<td>MEYER, Uwe</td>
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</tr>
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<td>MEYER, Xuan-Mi</td>
<td>P4.147</td>
</tr>
<tr>
<td>MEYNET, Nicolas</td>
<td>P3.196</td>
</tr>
<tr>
<td>MIANOWSKI, S.</td>
<td>P3.046, P3.048</td>
</tr>
<tr>
<td>MIANOWSKI, Slawomir</td>
<td>P3.047</td>
</tr>
<tr>
<td>MICCICHE, Gioacchino</td>
<td>P3.005, P3.007</td>
</tr>
<tr>
<td>MICHEL, Frederic</td>
<td>P1.138</td>
</tr>
<tr>
<td>MICHELETTI, Daniele</td>
<td>P1.033, P1.043</td>
</tr>
<tr>
<td>MIETELSKI, J.W.</td>
<td>P1.196</td>
</tr>
<tr>
<td>MIETELSKI, Jerzy</td>
<td>P3.001</td>
</tr>
<tr>
<td>MIGLIORI, Silvio</td>
<td>P4.158, P4.161</td>
</tr>
<tr>
<td>MIGUEL, Francisco</td>
<td>P4.029</td>
</tr>
<tr>
<td>MIKHALUK, Dmitry</td>
<td>P3.088</td>
</tr>
<tr>
<td>MIKLYAEV, Sergey</td>
<td>P3.087</td>
</tr>
<tr>
<td>MILANESIO, D.</td>
<td>I4.3</td>
</tr>
<tr>
<td>MILANESIO, Daniele</td>
<td>P2.007</td>
</tr>
<tr>
<td>MILANI, Francesco</td>
<td>P3.086, P3.087</td>
</tr>
<tr>
<td>MILITELLO, Fulvio</td>
<td>P1.082</td>
</tr>
<tr>
<td>MILLOT, Charles</td>
<td>P1.125</td>
</tr>
<tr>
<td>MILOCCO, Alberto</td>
<td>P1.066</td>
</tr>
<tr>
<td>MILOCCO, Alberto</td>
<td>O4A.1</td>
</tr>
<tr>
<td>MILWICH, Markus</td>
<td>P1.104</td>
</tr>
<tr>
<td>MIN, Kyung-Mi</td>
<td>P1.189, P3.165</td>
</tr>
<tr>
<td>MINUCCI, Simone</td>
<td>P2.011</td>
</tr>
<tr>
<td>MIRIZZI, Francesco</td>
<td>P2.007</td>
</tr>
<tr>
<td>MIRONES, Vicente</td>
<td>P2.092</td>
</tr>
<tr>
<td>MISHRA, Jyoti Shankar</td>
<td>P4.014</td>
</tr>
<tr>
<td>MISHRA, Jyotishankar</td>
<td>P4.163</td>
</tr>
<tr>
<td>MISSIRLIAN, M.</td>
<td>I5.2, P4.101</td>
</tr>
<tr>
<td>MISSIRLIAN, Marc</td>
<td>O2B.3, P1.070, P1.118, P3.123</td>
</tr>
<tr>
<td>MISTRANGELO, Chiara</td>
<td>P1.168, P2.151</td>
</tr>
<tr>
<td>MITARAI, Osamu</td>
<td>P1.046, P2.078</td>
</tr>
<tr>
<td>MITCHELL, John</td>
<td>P2.122</td>
</tr>
<tr>
<td>MITCHELL, N.</td>
<td>I3.1, P3.077</td>
</tr>
<tr>
<td>MITCHELL, Neil</td>
<td>P4.046</td>
</tr>
<tr>
<td>MITIN, Dmitry</td>
<td>P2.130</td>
</tr>
<tr>
<td>MITO, Toshiyuki</td>
<td>P1.004</td>
</tr>
<tr>
<td>MITTEAU, Raphael</td>
<td>P2.101, P2.102</td>
</tr>
<tr>
<td>MIYAMOTO, Mitsutaka</td>
<td>P4.120</td>
</tr>
<tr>
<td>MIYAZAWA, Junichi</td>
<td>P3.144</td>
</tr>
<tr>
<td>MIYAZAWA, Takeshi</td>
<td>P1.175, P1.176, P1.180</td>
</tr>
<tr>
<td>MIYOSHI, Yuya</td>
<td>P3.115</td>
</tr>
<tr>
<td>MOCHIZUKI, Jumpei</td>
<td>P3.148, P3.187, P3.188</td>
</tr>
<tr>
<td>MODESTOV, Victor</td>
<td>P2.132, P2.133, P2.137</td>
</tr>
<tr>
<td>MOELLER, C.</td>
<td>O2C.3</td>
</tr>
<tr>
<td>MOELLER, Charles</td>
<td>P2.037</td>
</tr>
<tr>
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<td>P3.192</td>
</tr>
<tr>
<td>MOLLA, Joaquin</td>
<td>P1.103</td>
</tr>
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<td>MOLLER, Soren</td>
<td>P3.034</td>
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<td>MONAKHOV, Igor</td>
<td>P4.036</td>
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<td>MONCADA, Victor</td>
<td>P1.069</td>
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<td>MONGE ALCAZAR, Miguel Angel</td>
<td>P2.182</td>
</tr>
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<td>ID</td>
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<td>P3.044</td>
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<td>P4.113</td>
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<td>P1.002, P1.003</td>
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<td>O2A.2, P1.190, P1.191</td>
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<td>P4.002, P4.004</td>
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<td>MOON, Se Yeon</td>
<td>P4.105</td>
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<td>P1.045</td>
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<td>P3.014</td>
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<td>O1A.1, P2.162, P3.146, P3.151</td>
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<td>P1.021, P1.023</td>
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<td>P1.028</td>
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<td>MORIMOTO, Junki</td>
<td>P3.186</td>
</tr>
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<td>P3.117</td>
</tr>
<tr>
<td>MORIN, Alexandre</td>
<td>P1.167</td>
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<tr>
<td>MORIOKA, Junya</td>
<td>P4.178</td>
</tr>
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<td>MORISADA, Yoshiaki</td>
<td>P1.178, P2.183</td>
</tr>
<tr>
<td>MORIYAMA, Shinichi</td>
<td>P1.033, P3.022</td>
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<td>MORO, Alessandro</td>
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<td>P1.128, P1.129, P2.004, P2.051</td>
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<td>O3B.3, P1.154, P1.155, P3.146, P3.172, P3.175</td>
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<td>MOROZOV, Anton</td>
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<td>P2.192, P2.194</td>
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<td>O1B.1, P1.082</td>
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<td>MOSCARDINI, Marigrazia</td>
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</tr>
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<td>MOSEEV, Dmitry</td>
<td>P4.124</td>
</tr>
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<td>O3A.4, P1.103, P3.203</td>
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<td>MUIR, David</td>
<td>P2.042, P4.062, P4.076</td>
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<td>MUKAI, Keisuke</td>
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<td>MUKHERJEE, A.</td>
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<td>MUKHERJEE, Aparajita</td>
<td>P4.034, P4.035</td>
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</tr>
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<td>MUKHERJEE, Samiran</td>
<td>P4.014, P4.163</td>
</tr>
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<td>MUKHIN, Eugene</td>
<td>P3.054, P4.050, P4.058</td>
</tr>
<tr>
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<td>P2.137</td>
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<tr>
<td>MUKTEPAVELA, Faina</td>
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<td>MUNOZ, Alejandro</td>
<td>P4.151</td>
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<td>MUNOZ, Angel</td>
<td>P1.101, P2.182</td>
</tr>
<tr>
<td>MURIKAMI, Haruyuki</td>
<td>P2.084, P2.085</td>
</tr>
<tr>
<td>MURARI, Andrea</td>
<td>P1.044, P3.045, P4.198</td>
</tr>
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<td>MURGATROYD, Julian</td>
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<tr>
<td>MURGATROYD, Julian</td>
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<td>P4.143</td>
</tr>
</tbody>
</table>
MURPHY, C. O2C.3
MURPHY, Chris J. P3.109
MUSALEK, Radomir P1.050
MUZI, Luigi P1.084, P1.086, P2.006
NA, Y.S. P1.079
NABARA, Yoshihiro O1C.1
NADASI, Gabor O3A.2, P1.152
NADEHARA, Kouji P1.004
NAGAOKA, Ken-Ichi P4.032
NAGASHIMA, Yoshihiko P1.046, P2.078
NAGATA, Takahiro P1.046, P2.078
NAGEL, Michael O5C.1, P1.009, P1.010
NAGY, A. O2C.3
NAGY, Daniel P3.053
NAGY, Domonkos Ferenc P3.053
NAISH, J. P3.046, P3.047, P4.169
NAISH, Jonathan P2.004, P4.199
NAISH, R. P3.047
NAISH, Richard P3.046
NAKAMICHI, Masaru P1.161, P3.166
NAKAMURA, Hirofumi P4.143
NAKAMURA, Kazuo P1.046, P1.073, P2.078
NAKANO, Haruhisa P4.032
NAKANO, Tomohide P2.019
NAKASHIMA, Hisatoshi P1.046, P2.078
NAKATA, Toshiya P1.181
NAM, Kwanwoo P1.127, P1.132
NAM, Kyoung P1.002, P1.003
NAM, Yong Un O1B.2
NAMKUNG, Won P2.032, P2.033
NANOBASHVILI, Irakli P1.018
NANOBASHVILI, Sulkhan P1.018, P2.060
NAOAKI, Yoshida P3.114
NAPPI, Eugenio P4.202
NARDON, Eric P1.069, P1.070
NARIS, S. O5A.1
NARUMI, Suzuki P4.189
NATSUME, Kyohei P1.138, P2.084
NAUJOKS, Dirk O5C.2, P3.038
Naulin, V. P1.017
Naulin, Volker P4.069
NAYAK, Pratik P4.014, P4.163
NAYLOR, Graham P1.048
NBTF TEAM AND THE CONTRIBUTING STAFF OF IO; F4E; JADA AND INDA
NEDZELSKIY, Igor P4.065
NEGRE, Llorenc P1.048
NEILSION, Hutch P2.154
NEILSON, George Hutch P3.201
NEILSON, Hutch P2.122
NEMCEK, Jiri P3.179
NEMOV, Aleksandr P2.136
NEMOV, Alexandr P2.137
NEMTSEV, Grigori P2.133
NERI, Carlo P3.044
NETO, Andre P1.048
NEU, R. I2.1
NEUBAUER, Olaf I5.1, P1.034, P2.067, P2.073
<table>
<thead>
<tr>
<th>Name</th>
<th>Page Numbers</th>
</tr>
</thead>
<tbody>
<tr>
<td>NEUBERGER, Heiko</td>
<td>O3B.2, P1.146, P1.152, P2.160</td>
</tr>
<tr>
<td>NEUNER, Ulrich</td>
<td>P2.069, P2.072</td>
</tr>
<tr>
<td>NEVEROV, Vladislav</td>
<td>P4.053</td>
</tr>
<tr>
<td>NEVIERE, Jean-Cristophe</td>
<td>P1.148</td>
</tr>
<tr>
<td>NEWMAN, Mark</td>
<td>O3B.1</td>
</tr>
<tr>
<td>NGO, Tran-Thanh</td>
<td>P1.069</td>
</tr>
<tr>
<td>NI, Muyi</td>
<td>P1.199</td>
</tr>
<tr>
<td>NICOLAI, Dirk</td>
<td>P2.067, P2.073</td>
</tr>
<tr>
<td>NICOLOLET, Sylvie</td>
<td>P1.088, P2.087, P2.089</td>
</tr>
<tr>
<td>NICULESCU, Alina</td>
<td>P4.155, P4.156</td>
</tr>
<tr>
<td>NIE, Baojie</td>
<td>P1.199</td>
</tr>
<tr>
<td>NIE, Miao</td>
<td>P3.194</td>
</tr>
<tr>
<td>NIE, Xingchen</td>
<td>P3.204</td>
</tr>
<tr>
<td>NIELSEN, A.H.</td>
<td>P1.017</td>
</tr>
<tr>
<td>NIKOLAEG, G.</td>
<td>P2.173</td>
</tr>
<tr>
<td>NIKOLIC, Vladica</td>
<td>P2.180</td>
</tr>
<tr>
<td>NIMAVAT, Hiren D.</td>
<td>P3.015</td>
</tr>
<tr>
<td>NISHIKIORI, Ryo</td>
<td>O2C.2</td>
</tr>
<tr>
<td>NISHITANI, Takeo</td>
<td>P3.071</td>
</tr>
<tr>
<td>NISHIYAMA, Koichi</td>
<td>P2.208</td>
</tr>
<tr>
<td>NOBLE, Craig</td>
<td>P4.036</td>
</tr>
<tr>
<td>NOBUTOKI, Minoru</td>
<td>P1.004</td>
</tr>
<tr>
<td>NOCENTE, M.</td>
<td>P3.046, P3.047</td>
</tr>
<tr>
<td>NOCENTE, Massimo</td>
<td>P2.080, P3.045, P3.066</td>
</tr>
<tr>
<td>NOCENTINI, Riccardo</td>
<td>P2.024, P2.025</td>
</tr>
<tr>
<td>NOGAMI, Shuhei</td>
<td>P1.179, P3.183</td>
</tr>
<tr>
<td>NOGUCHI, Mizuki</td>
<td>P4.115</td>
</tr>
<tr>
<td>NOGUCHI, Yuto</td>
<td>P4.128, P4.129, P4.133</td>
</tr>
<tr>
<td>NOH, Chang Hyun</td>
<td>P1.127, P1.132</td>
</tr>
<tr>
<td>NOMEN, Oriol</td>
<td>P3.008</td>
</tr>
<tr>
<td>NOMOTO, Kazuhiro</td>
<td>P2.085</td>
</tr>
<tr>
<td>NONHOFF, Marko</td>
<td>P4.067</td>
</tr>
<tr>
<td>NOONAN, Paul</td>
<td>P1.095</td>
</tr>
<tr>
<td>NORAJITRA, Prachai</td>
<td>P1.152, P2.160</td>
</tr>
<tr>
<td>NORRMAN, Sixten</td>
<td>P2.195</td>
</tr>
<tr>
<td>NOTERDAEME, J.-M.</td>
<td>I4.3</td>
</tr>
<tr>
<td>NOTERDAEME, Jean-Marie</td>
<td>P1.038, P3.033, P3.050, P4.020, P4.025, P4.086</td>
</tr>
<tr>
<td>NOTKIN, Gennadiy</td>
<td>P3.124</td>
</tr>
<tr>
<td>NOTO, Hiroyuki</td>
<td>P2.184</td>
</tr>
<tr>
<td>NOUAILLETAS, Remy</td>
<td>P1.069, P1.070, P2.044, P4.042</td>
</tr>
<tr>
<td>NOVAK KRMPOTIC, Sasa</td>
<td>P2.174</td>
</tr>
<tr>
<td>NOVAK, Sasa</td>
<td>P2.175</td>
</tr>
<tr>
<td>NOVELLO, Luca</td>
<td>P2.081, P2.082, P2.083</td>
</tr>
<tr>
<td>NOVOKSHENOV, Alexey</td>
<td>P2.136</td>
</tr>
<tr>
<td>NOWAK, Silvana</td>
<td>P1.043, P2.007</td>
</tr>
<tr>
<td>NOZAWA, Takashi</td>
<td>P1.177, P3.174, P4.111</td>
</tr>
<tr>
<td>NUNES, I.</td>
<td>I2.1</td>
</tr>
<tr>
<td>NUNIO, Francois</td>
<td>P1.088</td>
</tr>
<tr>
<td>NUSBAUM, Marc</td>
<td>P2.091</td>
</tr>
<tr>
<td>NYGREN, R.E.</td>
<td>O4B.3</td>
</tr>
<tr>
<td>NYGREN, Richard E.</td>
<td>P3.109</td>
</tr>
<tr>
<td>O'NEILL, R.</td>
<td>P4.057</td>
</tr>
<tr>
<td>OBERKOFLER, Martin</td>
<td>P4.121</td>
</tr>
<tr>
<td>OBRYK, Barbara</td>
<td>P2.004</td>
</tr>
<tr>
<td>OBUKHOV, Denis</td>
<td>P2.171</td>
</tr>
<tr>
<td>OCHIAI, Kentaro</td>
<td>P3.176, P3.178</td>
</tr>
<tr>
<td>OCHIAI, Ryouke</td>
<td>P1.180</td>
</tr>
<tr>
<td>OCHOA, S.</td>
<td>O5A.1</td>
</tr>
<tr>
<td>OCHOUKOV, R.</td>
<td>I4.3</td>
</tr>
<tr>
<td>ODA, Yasuhisa</td>
<td>P1.028, P3.023</td>
</tr>
<tr>
<td>ODA, Yasushi</td>
<td>P3.022</td>
</tr>
<tr>
<td>ODA, Yuki</td>
<td>P2.028</td>
</tr>
<tr>
<td>ODETTE, G. Robert</td>
<td>O3B.4</td>
</tr>
<tr>
<td>OFFERMANNS, Guido</td>
<td>P1.034, P2.073</td>
</tr>
</tbody>
</table>
PANAYOTIS, Stephanie O4B.1
PANAYOTOV, Dobromir O4C.3, P2.201
PANCHAL, Arun G. P3.015
PANCHAL, Manoj P1.127
PANCHAL, Paresh P4.014, P4.163
PANCHAL, Pradip N. P3.015
PANCHAL, Rohitkumar N. P3.015
PANDA, Nirmal P2.026
PANDYA, Bhargav P3.068
PANDYA, Darshit P3.062
PANDYA, Kaushal P4.013
PANEK, Radomir I2.3, P2.059, P2.061, P3.126
PANELLA, Maurizio P3.044
PANIN, Anatoly P1.090, P4.078
PANIZZA, Carlo P2.023
PANJAN, Matjaz P4.121
PANLUCCI, Francesco P1.023, P3.069, P4.009
PARAVASTU, Yuvakiran P3.015
PAREJA, Ramiro P1.101, P2.182
PAREKH, Tejas J. P3.015
PARIS, Peeter P4.066
PARK, Changho P3.174
PARK, Chul-Kyu P1.130
PARK, Dong-Seong P4.002, P4.004
PARK, Goon-Cherl P2.159
PARK, H. K. P4.090
PARK, H.K. P4.104
PARK, Hyeon K. O1B.2
PARK, Hyun Ki P2.117
PARK, Hyun Taek P4.001
PARK, Hyun Teak P4.005
PARK, Hyun-Ki P4.006, P4.159
PARK, Jihye P1.003
PARK, Jong Sung P1.202, P2.118, P2.120
PARK, JongSung P2.119
PARK, Jongyoon P1.080
PARK, Jun Young P1.191
PARK, Jun-Young O2A.2, P1.190
PARK, K. R. P4.090
PARK, Kaprai O1B.2, P2.116, P2.117, P4.001
PARK, Min P4.026, P4.027
PARK, S.H. P4.104
PARK, Seong Dae P1.164, P2.135, P4.106
PARK, Soo Hwan P2.117
PARK, Soo-Hwan P4.006, P4.159
PARK, Soon Chang P1.189
PARK, Sung Dae P4.105
PARK, Yi-Hyun P1.165, P1.188, P1.189, P1.192, P3.165
PARK, Youngjae P4.103
PARMAR, D. O1C.2
PARMAR, Kamubhai P4.013
PARMAR, Sanjay L. P3.068
PARSHUTIN, Evgeny P2.130
PARYS, Piotr P2.206
PASCAL, Jean-Yves P1.071
PASCAL, Romain P1.148
PASCUAL, Quentin P1.049
PASLER, Volker P3.145
PASQUALOTTO, Roberto P2.079
PASTOR, Carmen O3A.4
PASTOR, Ignacio P4.063
PASTOR, Jose Ignacio P1.104, P2.177
PATAKI, Adam O3A.2, P1.056
PATE, Anikumar P3.062
<table>
<thead>
<tr>
<th>Name</th>
<th>Code</th>
</tr>
</thead>
<tbody>
<tr>
<td>PATEL, A.</td>
<td>O1C.2</td>
</tr>
<tr>
<td>PATEL, Hiren</td>
<td>P1.006</td>
</tr>
<tr>
<td>PATEL, Hitesh Kumar K.</td>
<td>P2.026</td>
</tr>
<tr>
<td>PATEL, Hitesh S.</td>
<td>P3.015</td>
</tr>
<tr>
<td>PATEL, J.C.</td>
<td>P3.015</td>
</tr>
<tr>
<td>PATEL, Kaushal</td>
<td>P3.143</td>
</tr>
<tr>
<td>PATEL, Ketan G.</td>
<td>P3.015</td>
</tr>
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<td>P1.070</td>
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<tr>
<td>PATEL, Kirikumar B.</td>
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</tr>
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<td>P4.035</td>
</tr>
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</tr>
<tr>
<td>PATEL, Nirav</td>
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</tr>
<tr>
<td>PATEL, Nisarg</td>
<td>O1B.3</td>
</tr>
<tr>
<td>PATEL, Paresh. J.</td>
<td>P3.068</td>
</tr>
<tr>
<td>PATEL, Rakesh Kumar J.</td>
<td>P3.015</td>
</tr>
<tr>
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<td>P4.108</td>
</tr>
<tr>
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<td>P1.184</td>
</tr>
<tr>
<td>PATHAK, S.K.</td>
<td>P3.015</td>
</tr>
<tr>
<td>PATHAN, Firozkh S.</td>
<td>P3.015</td>
</tr>
<tr>
<td>PATTERSON, Michael</td>
<td>P4.018</td>
</tr>
<tr>
<td>PAUTASSO, Gabriella</td>
<td>P4.080</td>
</tr>
<tr>
<td>PAVEI, Mauro</td>
<td>P4.008</td>
</tr>
<tr>
<td>PAVEL, Pereslavtsev</td>
<td>P3.159</td>
</tr>
<tr>
<td>PAWLEY, Carl</td>
<td>P2.035</td>
</tr>
<tr>
<td>PAZ, Oscar</td>
<td>P4.077</td>
</tr>
<tr>
<td>PEACOCK, A.</td>
<td>I5.5</td>
</tr>
<tr>
<td>PEARCE, Robert</td>
<td>P1.131</td>
</tr>
<tr>
<td>PEDERSEN, Thomas Sunn</td>
<td>P2.064, P2.065</td>
</tr>
<tr>
<td>PEDRON, Diego</td>
<td>P1.022</td>
</tr>
<tr>
<td>PEGOURIE, B.</td>
<td>O5A.1</td>
</tr>
<tr>
<td>PEGOURIE, Bernard</td>
<td>P2.019, P4.161</td>
</tr>
<tr>
<td>PEGURIE, Bernard</td>
<td>P3.012</td>
</tr>
<tr>
<td>PEI, Kun</td>
<td>P1.121</td>
</tr>
<tr>
<td>PEI, Xiaofang</td>
<td>P2.042</td>
</tr>
<tr>
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<td>P1.114</td>
</tr>
<tr>
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<td>O3A.4</td>
</tr>
<tr>
<td>PENAFLORE, Benjamin G.</td>
<td>O5C.4</td>
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<td>PENALVA, Igor</td>
<td>P4.191</td>
</tr>
<tr>
<td>PENELIAU, Yannick</td>
<td>O3A.4</td>
</tr>
<tr>
<td>PENG, Changhong</td>
<td>P2.155</td>
</tr>
<tr>
<td>PENG, Jianfei</td>
<td>P4.095</td>
</tr>
<tr>
<td>PENG, Jiao</td>
<td>P2.191</td>
</tr>
<tr>
<td>PENG, Xuebing</td>
<td>P1.123</td>
</tr>
<tr>
<td>PENGCHENG, Long</td>
<td>P3.205</td>
</tr>
<tr>
<td>PENOT, Christophe</td>
<td>P2.048</td>
</tr>
<tr>
<td>PENZEL, Florian</td>
<td>O3A.2, P1.056, P1.057</td>
</tr>
<tr>
<td>PEREIRA, R. C.</td>
<td>P3.046, P3.048</td>
</tr>
<tr>
<td>PEREIRA, R.C.</td>
<td>P2.050</td>
</tr>
<tr>
<td>PEREIRA, Rita</td>
<td>O4A.3</td>
</tr>
<tr>
<td>PEREIRA, Rita C.</td>
<td>P3.045, P3.066, P4.086</td>
</tr>
<tr>
<td>PERESLAVTSEV, P.</td>
<td>I3.3</td>
</tr>
<tr>
<td>PERESLAVTSEV, Pavel</td>
<td>P1.152, P1.153, P1.201, P2.004</td>
</tr>
<tr>
<td>PEREZ, Albert</td>
<td>P3.028, P3.030</td>
</tr>
<tr>
<td>PEREZ, Alberto</td>
<td>P3.029, P3.067</td>
</tr>
<tr>
<td>PEREZ, German</td>
<td>O2B.1, P2.002</td>
</tr>
<tr>
<td>PEREZ-LASALA, Miguel</td>
<td>P3.052, P3.053</td>
</tr>
<tr>
<td>PERKINS, Rory</td>
<td>P3.035</td>
</tr>
<tr>
<td>PERN, Mauro</td>
<td>P2.023</td>
</tr>
<tr>
<td>PERSOEO, Valeria</td>
<td>P3.045</td>
</tr>
<tr>
<td>PERTSEV, Dmitrii</td>
<td>P2.171</td>
</tr>
<tr>
<td>PERUZZO, Simone</td>
<td>O1B.3, P1.075, P2.079</td>
</tr>
<tr>
<td>PESENTI, Paolo</td>
<td>P2.088</td>
</tr>
<tr>
<td>PESTCHANIEWS, Sergey</td>
<td>P2.115, P3.199</td>
</tr>
<tr>
<td>PETERKA, Matej</td>
<td>P3.126</td>
</tr>
</tbody>
</table>
PETERS, B.  
PETERS, Benedikt J.  
PETERSEN, Claus  
PETERSON, Per  
PETERSSON, Per  
PETIJEKIVS, Aleksandrs  
PFAFF, Eberhard  
PIEC, Z.  
PIETRO, Zito  
PIGATTO, Leonardo  
PIIP, Kaarel  
PILAN, Nicola  
PILARD, Vincent  
PILLON, M.  
PILOPP, Dirk  
PILOTTI, Riccardo  
PIMAZZONI, A.  
PIMAZZONI, Antonio  
PING, Wang  
PINGEL, Steffen  
PINNA, Tonio  
PINTSUK, G.  
PINTSUK, Gerald  
PIOVAN, R.  
PIOVESAN, Paolo  
PIPPAN, Reinhard  
PIRON, Chiara  
PIRON, Alfredo  
PIROS, Attila  
PISAREV, Alexander  
PIITTS, Richard  
PIITTS, Richard A  
PIJKOV, Andrei  
PIJKOV, Andrew  
PIZZO, Francesco  
PIZZUTO, Aldo  
PLATACIS, Ernests  
PLATANIA, Paola  
PLOCEK, Jiri  
PLOECKL, B.  
PLOECKL, Bernhard  
PLOYHAR, Steve  
PLUMMER, D.  
POCHEAU, Christine  
PODADERA, Ivan  
PODDA, Salvatore  
PODDUBNYI, Ivan  
POGGI, Luigi Antonio  
POHORECKI, Wladyslaw  
POITEVIN, Yves  
POKOL, G.I.  
POKOL, Gergo  
POLICARPO, Hugo  
POLLASTRONE, Fabio  
POLL, Gian Mario  
POLUMATKIN, Sergey  
PULUNOSKIY, Eduard  
POMARO, Nicola  
PONCE, Dan  
PONCET, Lionel  
PONKRATOV, Yuri  
PONKRATOV, Yuriy  
POPESCU, Bogdan  
POPKOV, Alexey
<table>
<thead>
<tr>
<th>Name</th>
<th>Refs</th>
</tr>
</thead>
<tbody>
<tr>
<td>RAMPAL, Gilles</td>
<td>P1.167</td>
</tr>
<tr>
<td>RANJAN, Rakesh</td>
<td>P1.006</td>
</tr>
<tr>
<td>RANJAN, Satupa</td>
<td>P1.069, P1.070</td>
</tr>
<tr>
<td>RANTALA, Seppo</td>
<td>P3.052</td>
</tr>
<tr>
<td>RAPISARDA, David</td>
<td>O1A.1, P2.160, P2.161, P2.162, P2.163, P3.119, P3.151</td>
</tr>
<tr>
<td>RAPP, Juergen</td>
<td>P4.113</td>
</tr>
<tr>
<td>RAPSON, Chris</td>
<td>P4.048</td>
</tr>
<tr>
<td>RAPSON, Christopher</td>
<td>P1.070, P4.043</td>
</tr>
<tr>
<td>RAPSON, Christopher J.</td>
<td>P1.039, P4.042</td>
</tr>
<tr>
<td>RAPSON, Christopher James</td>
<td>P4.047</td>
</tr>
<tr>
<td>RASINSKI, Marcin</td>
<td>O2B.2, O4A.4, P2.177, P3.180</td>
</tr>
<tr>
<td>RASMUSSEN, Jens Juul</td>
<td>P4.069</td>
</tr>
<tr>
<td>RASTOGI, Naveen</td>
<td>P4.130</td>
</tr>
<tr>
<td>RATHOD, Kulav</td>
<td>P3.143</td>
</tr>
<tr>
<td>RATHORE, Nisha Singh</td>
<td>P2.207</td>
</tr>
<tr>
<td>RATYNSKAIA, Svetlana</td>
<td>P3.126</td>
</tr>
<tr>
<td>RAUCH, J.</td>
<td>P2.035, P2.036</td>
</tr>
<tr>
<td>RAUPP, Gerhard</td>
<td>P1.070, P4.042, P4.043</td>
</tr>
<tr>
<td>RAVAL, B.</td>
<td>O1C.2</td>
</tr>
<tr>
<td>RAVAL, Dilip</td>
<td>P3.015</td>
</tr>
<tr>
<td>RAVAL, Jigar</td>
<td>P1.184</td>
</tr>
<tr>
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<td>P3.015</td>
</tr>
<tr>
<td>RAVENEL, Nathalie</td>
<td>P1.069, P1.070</td>
</tr>
<tr>
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<td>P4.024</td>
</tr>
<tr>
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<td>P2.007</td>
</tr>
<tr>
<td>RAZDOBARIN, Alexey</td>
<td>P4.058</td>
</tr>
<tr>
<td>RAZMEROV, Alexey</td>
<td>P2.106</td>
</tr>
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<td>O4B.2</td>
</tr>
<tr>
<td>REBAI, M.</td>
<td>P1.066</td>
</tr>
<tr>
<td>RECCIA, Luigi</td>
<td>P2.101</td>
</tr>
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<td>REICH, Matthias</td>
<td>P4.047</td>
</tr>
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<td>O3A.2, O3A.4, P1.055, P1.058, P1.128, P3.051</td>
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<td>REIMERDES, Holger</td>
<td>P1.082</td>
</tr>
<tr>
<td>REINECKE, Ernst-Arndt</td>
<td>P3.196</td>
</tr>
<tr>
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<td>O2B.4</td>
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<td>P4.157</td>
</tr>
<tr>
<td>RENARD, Sebastien</td>
<td>P1.012, P2.066</td>
</tr>
<tr>
<td>REUNGOAT, Mathieu</td>
<td>P1.147</td>
</tr>
<tr>
<td>REUTLINGER, Arnd</td>
<td>P4.055, P4.056</td>
</tr>
<tr>
<td>REY, Joerg</td>
<td>P1.146</td>
</tr>
<tr>
<td>REYNOLDS, S.</td>
<td>I5.5</td>
</tr>
<tr>
<td>REYNOLDS, Stephen</td>
<td>O3B.1</td>
</tr>
<tr>
<td>REZNICHENKO, Pavel</td>
<td>P4.159</td>
</tr>
<tr>
<td>RHEE, Chang-Kyu</td>
<td>P1.188</td>
</tr>
<tr>
<td>RIBEIRO, Isabel</td>
<td>O3C.4</td>
</tr>
<tr>
<td>RICAPITO, Italo</td>
<td>P1.148, P2.013, P2.014, P4.166</td>
</tr>
<tr>
<td>RICCARDO, Pedica</td>
<td>P3.202</td>
</tr>
<tr>
<td>RICCARDO, V.</td>
<td>P3.048</td>
</tr>
<tr>
<td>RICCARDO, Valeria</td>
<td>P3.046, P3.051, P4.119</td>
</tr>
<tr>
<td>RICCI, Daria</td>
<td>P1.033</td>
</tr>
<tr>
<td>RICHIUSA, Lorena</td>
<td>P2.146</td>
</tr>
<tr>
<td>RICHIUSA, Maria Lorena</td>
<td>P2.152, P3.005, P3.007</td>
</tr>
<tr>
<td>RICHOU, Marianne</td>
<td>O2B.3, P1.117, P1.118</td>
</tr>
<tr>
<td>RIEDL, Rudolf</td>
<td>P2.024, P2.025</td>
</tr>
<tr>
<td>RIEGO, Albert</td>
<td>P2.204</td>
</tr>
<tr>
<td>RIEMANN, Heike</td>
<td>P2.062, P3.036, P3.037</td>
</tr>
<tr>
<td>RIESCH, Johann</td>
<td>O2A.4, P3.112</td>
</tr>
<tr>
<td>RIETH, Michael</td>
<td>I4.1, P1.174, P3.131</td>
</tr>
<tr>
<td>RIETHM, Michael</td>
<td>O2A.4</td>
</tr>
<tr>
<td>RIGAMONTI, Davide</td>
<td>P3.045</td>
</tr>
<tr>
<td>RIGAMONTI, S.D.</td>
<td>P1.066</td>
</tr>
<tr>
<td>RIFFERS, Andre</td>
<td>P4.079</td>
</tr>
<tr>
<td>RIMINI, Fernanda</td>
<td>P1.044, P1.045, P2.003, P4.119</td>
</tr>
<tr>
<td>RINALDI, Luigi</td>
<td>P1.021, P3.082</td>
</tr>
<tr>
<td>Name</td>
<td>Code</td>
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<tr>
<td>RINCON, Esther</td>
<td>O1B.4</td>
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<tr>
<td>RIOS, Luis</td>
<td>O3A.4</td>
</tr>
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<td>RISPOLI, Natale</td>
<td>P1.043</td>
</tr>
<tr>
<td>RISSE, K.</td>
<td>P1.009</td>
</tr>
<tr>
<td>RISSE, Konrad</td>
<td>O5C.1</td>
</tr>
<tr>
<td>RITTICH, David</td>
<td>P1.037</td>
</tr>
<tr>
<td>RIVA, Giulio</td>
<td>P3.126</td>
</tr>
<tr>
<td>RIVA, M.</td>
<td>P2.050</td>
</tr>
<tr>
<td>RIVA, Marco</td>
<td>P1.061</td>
</tr>
<tr>
<td>RIVERO-RODRIGUEZ, Juan Francisco</td>
<td>P1.055</td>
</tr>
<tr>
<td>RIZZOLI, Andrea</td>
<td>P2.079</td>
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<td>ROBERTO, Pasqualotto</td>
<td>P2.080</td>
</tr>
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<td>ROBERTS, H.</td>
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</tr>
<tr>
<td>ROBERTS, Steve</td>
<td>O2A.1</td>
</tr>
<tr>
<td>ROBINSON, Stuart</td>
<td>P2.002</td>
</tr>
<tr>
<td>ROCA URGORRI, Fernando</td>
<td>P2.162</td>
</tr>
<tr>
<td>ROCCELLA, Selanna</td>
<td>O4B.2</td>
</tr>
<tr>
<td>ROCCHI, G.</td>
<td>P4.086</td>
</tr>
<tr>
<td>ROCCHI, Giuliano</td>
<td>P3.070</td>
</tr>
<tr>
<td>ROCCELLA, Riccardo</td>
<td>P3.056</td>
</tr>
<tr>
<td>ROCES, Jorge</td>
<td>P3.070</td>
</tr>
<tr>
<td>RODILLON, Damien</td>
<td>P1.136</td>
</tr>
<tr>
<td>RODIN, Igor</td>
<td>P2.171</td>
</tr>
<tr>
<td>RODRIGUES, Antonio</td>
<td>O4A.3</td>
</tr>
<tr>
<td>RODRIGUES, Antonio P.</td>
<td>P3.206</td>
</tr>
<tr>
<td>RODRIGUEZ, Alain</td>
<td>P4.185</td>
</tr>
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<td>RODRIGUEZ, Eduardo</td>
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</tr>
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<td>P4.138</td>
</tr>
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<td>RODRIGUEZ-RAMOS, Mauricio</td>
<td>P1.055</td>
</tr>
<tr>
<td>ROH, Byung-Ryul Roh</td>
<td>P1.130</td>
</tr>
<tr>
<td>ROHDE, Volker</td>
<td>P4.121</td>
</tr>
<tr>
<td>ROHOLLAHI, Akbar</td>
<td>P2.001</td>
</tr>
<tr>
<td>ROJO, Beatriz</td>
<td>P4.029</td>
</tr>
<tr>
<td>ROLLI, Rolf</td>
<td>P3.154</td>
</tr>
<tr>
<td>ROLLIG, M.</td>
<td>O5A.1</td>
</tr>
<tr>
<td>ROMAN, Guillermo</td>
<td>P4.077</td>
</tr>
<tr>
<td>ROMANIUK, Ryszard</td>
<td>P3.073</td>
</tr>
<tr>
<td>ROMANIKOV, Alexander</td>
<td>P2.106</td>
</tr>
<tr>
<td>ROMSY, Tomas</td>
<td>P3.011</td>
</tr>
<tr>
<td>RONDEN, Dennis</td>
<td>P3.018</td>
</tr>
<tr>
<td>RONGFEI, Wang</td>
<td>P1.068</td>
</tr>
<tr>
<td>ROSHAL, Alexander</td>
<td>P3.086</td>
</tr>
<tr>
<td>ROSINSKI, Marcin</td>
<td>P2.206</td>
</tr>
<tr>
<td>ROSS, John</td>
<td>P1.095</td>
</tr>
<tr>
<td>ROSSETTO, Federico</td>
<td>P4.007</td>
</tr>
<tr>
<td>ROSSI, Paolo</td>
<td>O4B.2</td>
</tr>
<tr>
<td>ROSSI, Riccardo</td>
<td>P4.196</td>
</tr>
<tr>
<td>ROSTOMASHVILI, George</td>
<td>P1.018</td>
</tr>
<tr>
<td>ROTT, Michael</td>
<td>O5C.3</td>
</tr>
<tr>
<td>ROTTI, Chandramouli</td>
<td>P2.026</td>
</tr>
<tr>
<td>ROZIER, Yoann</td>
<td>P1.029</td>
</tr>
<tr>
<td>ROZZIA, Davide</td>
<td>O1A.4</td>
</tr>
<tr>
<td>RUBEL, M.</td>
<td>I2.1</td>
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<tr>
<td>RUBEL, Marek</td>
<td>P3.034</td>
</tr>
<tr>
<td>RUCK, Sebastian</td>
<td>O3B.2</td>
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<tr>
<td>RUDAKOV, Dmitry</td>
<td>O4B.3</td>
</tr>
<tr>
<td>RUDAKOV, Dmitry L.</td>
<td>P3.109</td>
</tr>
<tr>
<td>RUDOVICA, Vita</td>
<td>P3.170</td>
</tr>
<tr>
<td>RUEDA, Fernando</td>
<td>P2.161</td>
</tr>
<tr>
<td>RUESS, Sebastian</td>
<td>O5B.3</td>
</tr>
<tr>
<td>RUI, Wang</td>
<td>P1.135</td>
</tr>
<tr>
<td>RUII, George</td>
<td>P2.176</td>
</tr>
<tr>
<td>RUIZ, Mariano</td>
<td>P3.060</td>
</tr>
<tr>
<td>RUMMEL, Kerstin</td>
<td>P2.122</td>
</tr>
</tbody>
</table>
RUMMEL, T. P1.009
RUMMEL, Thomas O5C.1, P1.010, P2.094
RUSEN, Cristian P3.113, P4.066
RUSTOMJI, Kaizad P2.034
RYAZANOV, Alexander P3.125
RYC, Leszek P2.069
RYDZY, Alexander P4.206
RYOSUKE, Seki P3.031
RZESNICKI, Tomasz O5B.3, P1.026, P3.025
SA, Jeong-Woo P1.130
SA, Rongyuan P2.202
SAARINEN, Hannu P3.052
SABOURIN, Flavien P2.126
SACHS, Edgar P3.083
SADAKOV, Sergey P2.130
SAFARIK, Pavel P3.011
SAFRONOV, Valery P2.104, P2.106, P2.115
SAGARA, Akio P1.110, P1.183, P2.172, P3.144, P3.186, P4.144
SAHEBI, Neda P2.203
SAIBENE, Gabriella P3.019, P3.020
SAIGUSA, Mikio P1.028
SAITO, Kenji P3.031
SAITO, Makiko P4.133
SAKAMOTO, Keishi P3.023
SAKAMOTO, Ryuichi P4.120
SAKAMOTO, Yoshiteru P1.041, P1.042, P1.115, P3.115, P3.137
SAKASEGAWA, Hideo P1.175, P1.183
SAKURADA, Shodai P3.173
SAKURAI, Shigeki P1.207
SAKURAI, Shinji P1.073, P2.019
SALAZAR, E. P3.081
SALVATORE, Marocco P4.088
SAMAILE, F. I5.2
SAMAILE, Franck P3.123
SAMAILE, Frank O4B.3, P1.070, P1.136
SAMANIEGO, Fernando P2.103, P2.185
SAMEC, Karel P2.107
SAMSONOV, Dmitrii P2.137
SAMSONOV, Dmitry P4.058
SANCHEZ, Emilio P4.029
SANCHEZ, Fernando P3.172, P3.175
SANCHEZ, J. I3.2
SANCHEZ, Maria P2.188
SANCHEIS-SANCHEZ, Lucia P1.055
SANDFORD, Guy P1.049
SANG, Ge P3.162
SANTOS SILVA, Phillip P3.019, P3.020, P3.021
SANTOS, B. P2.050, P3.046, P3.047
SANTOS, Bruno O4A.3, P3.063, P3.064, P3.065, P3.206
SANTOS, Goncalo P4.048
SANTOS, Jorge P4.086
SANTOS, Jorge M. P4.048
SANTRA, Prosenjit P3.015
SANTRAINE, Benjamin P1.069
SANTUCCI, A. O5A.1
SANTUCCI, Alessia P4.148, P4.149, P4.150
SANZ, J. I3.3
SANZ, Javier P1.200, P1.201, P2.142, P3.203
SARKAR, Biswanath P3.082
SARRIONANDIA-IBARRA, A. P4.191
SARTORI, E. O5A.1
SARTORI, Emanuele P2.021, P2.022
SARTORI, Filippo P1.023, P1.048, P3.069
SEON, Changrae
SEONG, Taesik
SERGIENKO, G.
SERGIENKO, Gennady
SERGIS, Antonis
SERIANNI, G.
SERIANNI, Gianluigi
SERIKOV, Arkady
SERIO, Luigi
SERIZAWA, Hisashi
SESTAK, David
SESTAN, Andreja
SEYVET, Fabien
SJOBBA, Stefano
SHABBIR, Aqsa
SHABLONIN, Jevgeni
SHAGNIEV, Oleg
SHAH, Pankil R.
SHANG, Leiming
SHARAFUTDINOV, Marat
SHARMA, Aashoo
SHARMA, Atish L.
SHARMA, Dinesh Kumar
SHARMA, L. K.
SHARMA, Lalit
SHARMA, Manika
SHARMA, Rajiv
SHARMA, Ridhima
SHARMA, Sanjeev K.
SHATIL, Nicolai
SHAW, R.
SHEKHTMAN, Lev
SHELUKHIN, Dmitry
SHEPHERD, Alastair
SHI, Bo
SHI, Shanshuang
SHI, Y.J.
SHI, Yue-Jiang
SHI, Z.B.
SHI, Zongqian
SHIBAMA, Yusuke
SHIBATA, Naoki
SHIBUYA, Masayuki
SHIKHOTSEV, Igor
SHIKHOTSEV, Igor V.
SHILOV, Alexander
SHIM, Hee-Jin
SHIM, Heejin
SHIMADA, Katsuhiro
SHIMIZU, Katsuhiro
SHIMWELL, Jonathan
SHIN, Chang Ho
SHIN-MURA, Kiyoto
SHIRAI, H.
SHMAKOV, Alexander
SHODAI, Sakurada
SHOSHIN, Andrey
SHUFF, Robin
SHUICHI, Saito
SHUJI, Kamio
SHUKLA, Braj
SHUKLA, Braj Kishor
SHYAM, Anurag
SIBILIA, Marc
<table>
<thead>
<tr>
<th>Name</th>
<th>Pages</th>
</tr>
</thead>
<tbody>
<tr>
<td>SIBOIS, Romain</td>
<td>P3.142</td>
</tr>
<tr>
<td>SIBURNE, Scott</td>
<td>P4.061</td>
</tr>
<tr>
<td>SIEBER, Thomas</td>
<td>P2.071</td>
</tr>
<tr>
<td>SIEFKEN, Udo</td>
<td>P1.104</td>
</tr>
<tr>
<td>SIEGLIN, Bernhard</td>
<td>O5C.3</td>
</tr>
<tr>
<td>SIGNOR ET, Jacqueline</td>
<td>P1.070</td>
</tr>
<tr>
<td>SILBURN, Scott</td>
<td>OA4A.1, P4.119</td>
</tr>
<tr>
<td>SILVA, A.</td>
<td>I4.3</td>
</tr>
<tr>
<td>SILVA, Antonio</td>
<td>P1.052, P2.060, P4.048, P4.086</td>
</tr>
<tr>
<td>SILVA, Joao</td>
<td>P1.138</td>
</tr>
<tr>
<td>SILVA, Miguel</td>
<td>P3.029, P3.067</td>
</tr>
<tr>
<td>SIMIONATO, Paola</td>
<td>P1.021, P3.069, P4.009</td>
</tr>
<tr>
<td>SIMON, Muriel</td>
<td>P1.021, P1.022, P2.023, P3.082, P3.083</td>
</tr>
<tr>
<td>SIMONETTO, Alessandro</td>
<td>P3.075</td>
</tr>
<tr>
<td>SIMONIN, Alain</td>
<td>P4.020</td>
</tr>
<tr>
<td>SIMROCK, Stefan</td>
<td>P1.048, P4.059</td>
</tr>
<tr>
<td>SINGH, Aditya</td>
<td>P1.001, P1.006</td>
</tr>
<tr>
<td>SINGH, Akhilesh</td>
<td>P3.015</td>
</tr>
<tr>
<td>SINGH, Mahendrajit</td>
<td>P4.008</td>
</tr>
<tr>
<td>SINGH, N. P.</td>
<td>O1C.2</td>
</tr>
<tr>
<td>SINGH, Narinder Pal</td>
<td>P3.062</td>
</tr>
<tr>
<td>SINGH, R.</td>
<td>O1C.2</td>
</tr>
<tr>
<td>SINGH, Raghuraj</td>
<td>P4.034</td>
</tr>
<tr>
<td>SIPS, A.</td>
<td>P4.169</td>
</tr>
<tr>
<td>SIPS, Adrianus</td>
<td>P2.003</td>
</tr>
<tr>
<td>SIPS, Adriamus C. C.</td>
<td>P1.045, P4.170</td>
</tr>
<tr>
<td>SIRAGUSA, Marco</td>
<td>O1B.3</td>
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<tr>
<td>SIRAVO, Ugo</td>
<td>P3.024, P3.027, P3.028, P3.029, P3.030, P3.067</td>
</tr>
<tr>
<td>SIRINELLI, Antoine</td>
<td>P4.124</td>
</tr>
<tr>
<td>SIROTTI, Fabien</td>
<td>P4.009</td>
</tr>
<tr>
<td>SISKA, Filip</td>
<td>P3.190, P3.191</td>
</tr>
<tr>
<td>SISTLA, Sree</td>
<td>O2A.4</td>
</tr>
<tr>
<td>SITA, Luca</td>
<td>P1.021, P3.082</td>
</tr>
<tr>
<td>SIKAKOV, Mazhyn</td>
<td>P1.109, P4.116</td>
</tr>
<tr>
<td>SKILTON, Robert</td>
<td>P3.138</td>
</tr>
<tr>
<td>SKILTON, Robert M.</td>
<td>P4.138</td>
</tr>
<tr>
<td>SKLADNOV, Konstantin</td>
<td>P2.130</td>
</tr>
<tr>
<td>SKODA, Radek</td>
<td>P2.111</td>
</tr>
<tr>
<td>SKOVORODIN, Dmitry</td>
<td>P3.125</td>
</tr>
<tr>
<td>SLADEK, Petr</td>
<td>P1.051</td>
</tr>
<tr>
<td>SMID, Miroslav</td>
<td>P3.191</td>
</tr>
<tr>
<td>SMILEY, Matthew</td>
<td>P4.057, P4.079</td>
</tr>
<tr>
<td>SMIRNOV, A.</td>
<td>P4.038</td>
</tr>
<tr>
<td>SMIRNOV, Alexander</td>
<td>P2.133</td>
</tr>
<tr>
<td>SMITH, George</td>
<td>O2A.1</td>
</tr>
<tr>
<td>SMITH, George D. W.</td>
<td>P3.107</td>
</tr>
<tr>
<td>SMITH, J.</td>
<td>P3.081</td>
</tr>
<tr>
<td>SMITH, Keith</td>
<td>P4.114</td>
</tr>
<tr>
<td>SMITH, M.</td>
<td>P4.057</td>
</tr>
<tr>
<td>SMITH, Mark</td>
<td>P3.054, P4.079</td>
</tr>
<tr>
<td>SNEAD, Lance L.</td>
<td>P1.106</td>
</tr>
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<td>SNIPES, Joe</td>
<td>P4.044</td>
</tr>
<tr>
<td>SNIPES, Joseph Alan</td>
<td>P4.046</td>
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<td>SNOJ, L.</td>
<td>P1.066</td>
</tr>
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<td>SO, Maruyama</td>
<td>P1.008</td>
</tr>
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<td>SOARE, S.</td>
<td>P3.047</td>
</tr>
<tr>
<td>SOARE, Sorin</td>
<td>P3.046, P3.048</td>
</tr>
<tr>
<td>SOBRENO, Giulia</td>
<td>P3.197</td>
</tr>
<tr>
<td>SOFIILCA, Nicolae</td>
<td>P4.156</td>
</tr>
<tr>
<td>SOKOLOV, Michail</td>
<td>P2.045</td>
</tr>
<tr>
<td>SOLETO, Alfonso</td>
<td>O1B.4, P2.018, P3.075, P4.029</td>
</tr>
<tr>
<td>SOLOKHA, Vladimir</td>
<td>P4.058</td>
</tr>
<tr>
<td>SOLOMATIN, Roman</td>
<td>P3.124</td>
</tr>
<tr>
<td>SOLOPEKO, Alexander</td>
<td>P1.094</td>
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<tr>
<td>Name</td>
<td>Session(s)</td>
</tr>
<tr>
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<td>P4.058</td>
</tr>
<tr>
<td>SOMEYA, Youji</td>
<td>P1.042, P3.115, P3.133, P3.137</td>
</tr>
<tr>
<td>SON, Soo-hyun</td>
<td>P2.075</td>
</tr>
<tr>
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<td>P3.015</td>
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<tr>
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<td>O5A.1</td>
</tr>
<tr>
<td>SONATO, Piergiorgio</td>
<td>O1B.3, P2.007, P2.144, P4.020, P4.021</td>
</tr>
<tr>
<td>SONG, Gang</td>
<td>O4C.1</td>
</tr>
<tr>
<td>SONG, Inho</td>
<td>P3.087, P4.046</td>
</tr>
<tr>
<td>SONG, J. H.</td>
<td>P4.090</td>
</tr>
<tr>
<td>SONG, J.H.</td>
<td>P4.104</td>
</tr>
<tr>
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<td>P2.117, P4.006</td>
</tr>
<tr>
<td>SONG, Jing</td>
<td>P2.015, P2.016, P3.207</td>
</tr>
<tr>
<td>SONG, Nak Hyoun</td>
<td>P4.001</td>
</tr>
<tr>
<td>SONG, Nak-Hyung</td>
<td>P4.004</td>
</tr>
<tr>
<td>SONG, Nak-hyung</td>
<td>P4.002</td>
</tr>
<tr>
<td>SONG, Xianming</td>
<td>P4.049</td>
</tr>
<tr>
<td>SONG, Yong</td>
<td>O4C.1</td>
</tr>
<tr>
<td>SONG, Zhiquan</td>
<td>P3.076</td>
</tr>
<tr>
<td>SONI, D.</td>
<td>O1C.2</td>
</tr>
<tr>
<td>SONI, Jignesh</td>
<td>P4.013</td>
</tr>
<tr>
<td>SONI, Jigneshkumar</td>
<td>P1.070</td>
</tr>
<tr>
<td>SOROKIN, Aleksey V.</td>
<td>P3.030</td>
</tr>
<tr>
<td>SOTO, Carlota</td>
<td>O3B.3</td>
</tr>
<tr>
<td>SOUSA, J.</td>
<td>P2.050</td>
</tr>
<tr>
<td>SOUSA, Jorge</td>
<td>O4A.3, P1.048, P3.063, P3.064, P3.065, P3.066, P3.206</td>
</tr>
<tr>
<td>SOZZI, Carlo</td>
<td>P1.033, P1.043, P1.045, P2.007</td>
</tr>
<tr>
<td>SPAEH, Peter</td>
<td>P3.026</td>
</tr>
<tr>
<td>SPAGNUOLO, Gandolfo Alessandro</td>
<td>P1.145, P3.130</td>
</tr>
<tr>
<td>SPASSOVSKY, Ivan</td>
<td>P4.023, P4.024</td>
</tr>
<tr>
<td>SPECOGNA, Ruben</td>
<td>P2.040, P2.041, P4.045</td>
</tr>
<tr>
<td>SPIGO, Giancarlo</td>
<td>P4.202</td>
</tr>
<tr>
<td>SPITSYN, Alexander</td>
<td>P1.150</td>
</tr>
<tr>
<td>SPOLAORE, Monica</td>
<td>P4.069</td>
</tr>
<tr>
<td>SPRING, Anett</td>
<td>P2.062, P3.036, P3.037</td>
</tr>
<tr>
<td>SPRING, Annette</td>
<td>P1.070</td>
</tr>
<tr>
<td>SRIDHAR, B. V. S. N. N. P.</td>
<td>P3.068</td>
</tr>
<tr>
<td>SRIKANTH, G.</td>
<td>P3.015</td>
</tr>
<tr>
<td>SRINIVASAN, R.</td>
<td>P1.092</td>
</tr>
<tr>
<td>STABLES, G.</td>
<td>P4.169</td>
</tr>
<tr>
<td>STADLER, Reinhold</td>
<td>P2.123, P2.124, P2.125</td>
</tr>
<tr>
<td>STAMATELATOS, I. E.</td>
<td>P1.196</td>
</tr>
<tr>
<td>STAMATELATOS, Ion E.</td>
<td>P2.004</td>
</tr>
<tr>
<td>STANCAR, Z.</td>
<td>P3.048</td>
</tr>
<tr>
<td>STANCU, Cristian</td>
<td>P2.176, P3.108</td>
</tr>
<tr>
<td>STANGEBY, P.C.</td>
<td>O4B.3</td>
</tr>
<tr>
<td>STANGEBY, Peter C.</td>
<td>P3.109</td>
</tr>
<tr>
<td>STANKUNAS, G.</td>
<td>I3.3</td>
</tr>
<tr>
<td>STANKUNAS, Gedelminas</td>
<td>P1.201, P1.203</td>
</tr>
<tr>
<td>STARACE, Fabio</td>
<td>P2.010, P2.011</td>
</tr>
<tr>
<td>STARZ, Ronald</td>
<td>P4.069</td>
</tr>
<tr>
<td>STEFFEN, Paul-Martin</td>
<td>P3.196</td>
</tr>
<tr>
<td>STEPAKEK, Jan</td>
<td>P2.110, P2.111, P2.200</td>
</tr>
<tr>
<td>STEFANOV, Boris</td>
<td>P1.085, P1.089</td>
</tr>
<tr>
<td>STEPHEN, Adam</td>
<td>P1.048</td>
</tr>
<tr>
<td>STEPHEN, Manoosh</td>
<td>P4.130</td>
</tr>
<tr>
<td>STIEGLITZ, Robert</td>
<td>P1.197</td>
</tr>
<tr>
<td>STIJKEL, M.P.</td>
<td>P3.161</td>
</tr>
<tr>
<td>STOBER, Joerg</td>
<td>P4.047</td>
</tr>
<tr>
<td>STOBER, Jorg</td>
<td>P1.038</td>
</tr>
<tr>
<td>STOCKEL, Jan</td>
<td>P2.057</td>
</tr>
<tr>
<td>STOKLASA, Jaroslav</td>
<td>P1.205</td>
</tr>
<tr>
<td>STRATIL, Ludek</td>
<td>P3.190, P3.191</td>
</tr>
<tr>
<td>Name</td>
<td>Pagenumbers</td>
</tr>
<tr>
<td>--------------------</td>
<td>---------------------</td>
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<tr>
<td>STRATTON, B.</td>
<td>P4.057</td>
</tr>
<tr>
<td>STRATTON, Brentley</td>
<td>P4.079</td>
</tr>
<tr>
<td>STRAUSS, Dirk</td>
<td>O5B.3, P1.054, P3.026</td>
</tr>
<tr>
<td>STREBKOV, Yuri</td>
<td>P2.104, P2.106, P2.130</td>
</tr>
<tr>
<td>STRITTMATTER, Tobias</td>
<td>P3.024</td>
</tr>
<tr>
<td>STROBEL, H.</td>
<td>O5A.1</td>
</tr>
<tr>
<td>STUPAR, Dusan</td>
<td>P2.027</td>
</tr>
<tr>
<td>STUPKA, Petr</td>
<td>P4.188</td>
</tr>
<tr>
<td>SUAREZ, Daniel</td>
<td>P2.164</td>
</tr>
<tr>
<td>SUBBOTIN, Mikhail</td>
<td>P4.016, P4.202</td>
</tr>
<tr>
<td>SUBRT, Jan</td>
<td>P3.179</td>
</tr>
<tr>
<td>SUDKI, Bassem</td>
<td>P3.019, P3.020, P3.021</td>
</tr>
<tr>
<td>SUEOKA, Michiharu</td>
<td>P1.046</td>
</tr>
<tr>
<td>SUGANDHI, Ritesh</td>
<td>P1.070</td>
</tr>
<tr>
<td>SUGIYAMA, Taishi</td>
<td>P3.121</td>
</tr>
<tr>
<td>SUMOD, C. B.</td>
<td>P3.068</td>
</tr>
<tr>
<td>SUN, Guangyao</td>
<td>P2.015, P2.016, P3.207</td>
</tr>
<tr>
<td>SUN, Lujun</td>
<td>P3.009</td>
</tr>
<tr>
<td>SUN, Ming</td>
<td>P3.193</td>
</tr>
<tr>
<td>SUN, Tengfei</td>
<td>P2.138</td>
</tr>
<tr>
<td>SUPE, Armsi</td>
<td>P3.170</td>
</tr>
<tr>
<td>SURREY, E.</td>
<td>P3.077</td>
</tr>
<tr>
<td>SURREY, Elizabeth</td>
<td>O5A.2, P1.082</td>
</tr>
<tr>
<td>SUTHAR, Gajendra</td>
<td>P4.034</td>
</tr>
<tr>
<td>SUTTROP, Wolfgang</td>
<td>P4.091, P4.093</td>
</tr>
<tr>
<td>SUZUKI, T.</td>
<td>P1.074</td>
</tr>
<tr>
<td>SUZUKI, Takumi</td>
<td>P4.120, P4.143</td>
</tr>
<tr>
<td>SUZUKI, Tsuneo</td>
<td>P1.020</td>
</tr>
<tr>
<td>SVEINSSON, Lennart</td>
<td>P1.023, P2.020, P3.069, P4.009, P4.046</td>
</tr>
<tr>
<td>SVIRIDENKO, Maksim</td>
<td>P2.104</td>
</tr>
<tr>
<td>SVIRIDENKO, Maxim</td>
<td>P2.106</td>
</tr>
<tr>
<td>SZOBOSLA, Vojtech</td>
<td>P2.001</td>
</tr>
<tr>
<td>SWATTON, Emily</td>
<td>P2.002</td>
</tr>
<tr>
<td>SYCHEVA, Svetlana</td>
<td>P2.136</td>
</tr>
<tr>
<td>SYKES, A.</td>
<td>P1.047</td>
</tr>
<tr>
<td>SYKES, Alan</td>
<td>P1.095</td>
</tr>
<tr>
<td>SYTCHEVSKY, Sergei</td>
<td>P4.010, P4.010, P4.011, P4.012</td>
</tr>
<tr>
<td>SZABOLICS, Tamas</td>
<td>P2.068</td>
</tr>
<tr>
<td>SZALARDY, Sandor</td>
<td>P1.125</td>
</tr>
<tr>
<td>SZEPESI, Tamás</td>
<td>P2.064, P2.068</td>
</tr>
<tr>
<td>SZUTYANYI, Mark</td>
<td>P2.061</td>
</tr>
<tr>
<td>SZYDROWSKI, Adam</td>
<td>P3.047</td>
</tr>
<tr>
<td>TACCONELLI, Massimiliano</td>
<td>P2.086</td>
</tr>
<tr>
<td>TADA, Naoya</td>
<td>P1.181</td>
</tr>
<tr>
<td>TADDIA, Giuseppe</td>
<td>P1.021, P2.082, P3.082</td>
</tr>
<tr>
<td>TAEGU, Lee</td>
<td>P3.059</td>
</tr>
<tr>
<td>TAGUCHI, Akira</td>
<td>P4.115</td>
</tr>
<tr>
<td>TAK, Tachyun</td>
<td>P3.202</td>
</tr>
<tr>
<td>TAKADA, Suguru</td>
<td>P1.004</td>
</tr>
<tr>
<td>TAKAHASHI, Koji</td>
<td>P1.028, P3.022, P3.023</td>
</tr>
<tr>
<td>TAKAHATA, Kazuya</td>
<td>P1.004</td>
</tr>
<tr>
<td>TAKASE, Haruhiko</td>
<td>P1.041, P3.115, P3.137</td>
</tr>
<tr>
<td>TAKASHI, Mutoh</td>
<td>P3.031</td>
</tr>
<tr>
<td>TAKAYAMA, Sadatsugu</td>
<td>P4.144</td>
</tr>
<tr>
<td>TAKECHI, Manabu</td>
<td>P1.046, P1.073</td>
</tr>
<tr>
<td>TAKEDA, Nobukazu</td>
<td>P4.128, P4.129, P4.133</td>
</tr>
<tr>
<td>TAKEDA, Shutaro</td>
<td>P1.207</td>
</tr>
<tr>
<td>TAKEI, Yasuhiko</td>
<td>P4.032</td>
</tr>
<tr>
<td>TAKEISHI, Toshiharu</td>
<td>P1.206, P3.167</td>
</tr>
<tr>
<td>TAKIMOTO, Toshikio</td>
<td>P4.031, P4.112, P4.118</td>
</tr>
<tr>
<td>TAKUMI, Chikada</td>
<td>P3.114</td>
</tr>
<tr>
<td>TALIERCIO, Cesare</td>
<td>P1.021, P1.023, P2.038, P3.069</td>
</tr>
<tr>
<td>TAMAGNONE, Michele</td>
<td>P1.025</td>
</tr>
<tr>
<td>TAMULEVICIUS, Sigita</td>
<td>P3.170</td>
</tr>
</tbody>
</table>
TIGELIS, Ioannis O5B.3, P3.025
TIKKA, Petri P3.052
TITUS, Peter P2.122, P2.154, P2.167, P3.134, P3.158
TOBARI, Hiroyuki O2C.2, P2.028
TOBITA, Kenji P1.041, P1.042, P1.115, P3.115, P3.133, P3.137
TODD, N. Tom O5A.2
TOIGO, V. I4.2
TOIGO, Vanni P1.021, P1.022, P2.011, P3.083
TOJO, Hiroshi P1.072
TOKITANI, Masayuki P1.110, P4.120
TOKUNAGA, Kazutoshi P1.046, P2.078
TOKUNAGA, Shinsuke P1.041, P1.042, P1.115
TOLIAS, Panagiotis P3.126
TOLLIN, Marco P2.079
TOLOCHKO, Boris P3.125
TOLSTYAKOV, Sergei P2.137
TOLSTYAKOV, Sergey P4.058
TOMARCHIO, V. I1.3
TOMARCHIO, Valerio P2.086, P2.088
TOMASSETTI, Giordano P2.006
TOMES, Matej P3.126
TOMILOV, Sergey P2.104
TOMOAKI, Kunugi P1.156
TONEGAWA, Akira P4.031, P4.118
TOOKER, Joseph O2C.3
TORIKAI, Yuji P4.115
TORRE, Alexandre O1C.3, P1.088, P2.087, P2.089, P2.091
TURREBLANCA, H. O2C.3, P2.036
TURREZAN, Antonio P2.037
TOSTI, S. O5A.1
TOSTI, Silvano P4.148, P4.149, P4.150
TOUSSAINT, Matthieu P3.027, P3.030
TOYAMA, Takeshi P1.179
TRACZ, Grzegorz P2.053
TRAN, Minh Quang O5B.3, P3.025, P4.019, P4.020
TRAVERE, Jean Marcel O3A.4
TRAVERE, Jean-Marcel P1.069, P1.070
TRAVLEEV, A. I3.3
TRAVLEEV, Anton P1.128, P1.129, P1.201
TREUTERER, Wolfgang P1.070
TREUTTERER, Wolfgang P1.039, P4.042, P4.043, P4.047, P4.048
TRIPATHI, Sudhir P4.108
TRIPATHI, V.S. P4.163
TRIPSKI, Matej P3.034
TRIVEDI, R.G. O1C.2, P4.034
TRIVEDI, Rajesh P4.035
TROISNE, Marc P1.171
TROULAY, Michele P4.147
TRUNEV, Yuriy P3.125
TSIRONIS, Christos P1.030
TSITRONE, E. I5.2
TSITRONE, Emanuelle O2B.3
TSUCHIYA, Katsuhiko P2.084, P2.085
TSUMORI, Katsuyoshi P4.032
TUCCILLO, A. A. I4.3
TUCCILLO, Angelo A. P4.086
TUCCILLO, Angelo Antonio P2.007
TUDISCO, O. I4.3
TUDISCO, Onofrio P3.043, P4.086
TURETTA, Andrea P1.025
TURNER, A. I3.3
TURNER, Andrew P1.129, P2.131
TURNER, Ingrid P2.002
TURTU', Simonetta P1.084
<table>
<thead>
<tr>
<th>Authors</th>
<th>Pages</th>
</tr>
</thead>
<tbody>
<tr>
<td>VARMORA, Pankaj</td>
<td>P3.015</td>
</tr>
<tr>
<td>VAROUTIS, S.</td>
<td>O5A.1</td>
</tr>
<tr>
<td>VASILEIADIS, Nikos</td>
<td>P4.160</td>
</tr>
<tr>
<td>VASILEIADOU, Soultana</td>
<td>P1.030</td>
</tr>
<tr>
<td>VASILEV, Alexandr S.</td>
<td>P2.056</td>
</tr>
<tr>
<td>VASILIEV, Vyacheslav</td>
<td>P4.012</td>
</tr>
<tr>
<td>VASULOPOULOU, T.</td>
<td>P1.196</td>
</tr>
<tr>
<td>VASULOPOULOU, Theodora</td>
<td>P2.004</td>
</tr>
<tr>
<td>VASILYEV, Alexander</td>
<td>P3.125</td>
</tr>
<tr>
<td>VASINA, Yana</td>
<td>P4.117</td>
</tr>
<tr>
<td>VASQUEZ, J.</td>
<td>P4.057</td>
</tr>
<tr>
<td>VAYAKIS, George</td>
<td>P1.048, P1.049, P1.050, P1.051, P1.055, P1.058, P4.046, P4.050, P4.124</td>
</tr>
<tr>
<td>VEGA, Jesus</td>
<td>P1.045, P4.063</td>
</tr>
<tr>
<td>VELTRI, Pierluigi</td>
<td>P2.021</td>
</tr>
<tr>
<td>VENTURA, Rodrigo</td>
<td>O3C.4, P4.135</td>
</tr>
<tr>
<td>VENTURINI, Alessandro</td>
<td>P2.012, P2.014</td>
</tr>
<tr>
<td>VERDINI, Luigi</td>
<td>O4B.2</td>
</tr>
<tr>
<td>VERDOOLAEGE, Geert</td>
<td>P3.050</td>
</tr>
<tr>
<td>VERDU, Marina</td>
<td>P3.175</td>
</tr>
<tr>
<td>VERES, Gabor</td>
<td>P2.186, P3.053, P3.072</td>
</tr>
<tr>
<td>VERGARA, Antonio</td>
<td>P4.046</td>
</tr>
<tr>
<td>VERGER, Jean-Marc</td>
<td>O2C.1</td>
</tr>
<tr>
<td>VERHOEFF, Peter</td>
<td>P4.079</td>
</tr>
<tr>
<td>VERLAAN, A.L.</td>
<td>P4.057</td>
</tr>
<tr>
<td>VERLAAN, Ad</td>
<td>P4.079</td>
</tr>
<tr>
<td>VERSHIKOV, Vladimir</td>
<td>P2.136</td>
</tr>
<tr>
<td>VEVERIER, Michel</td>
<td>P1.034</td>
</tr>
<tr>
<td>VESELY, Ladislav</td>
<td>P2.110, P2.197</td>
</tr>
<tr>
<td>VESHCHEV, Evgeny</td>
<td>P1.055</td>
</tr>
<tr>
<td>VIAN, Dionisio</td>
<td>P2.081</td>
</tr>
<tr>
<td>VIANELLO, Nicola</td>
<td>P4.069</td>
</tr>
<tr>
<td>VICIAN, Martin</td>
<td>P1.147</td>
</tr>
<tr>
<td>VIERERBL, L.</td>
<td>P4.073</td>
</tr>
<tr>
<td>VIERERBL, Ladislav</td>
<td>P1.050</td>
</tr>
<tr>
<td>VIERLE, T.</td>
<td>I4.3</td>
</tr>
<tr>
<td>VIK, Ondrej</td>
<td>P3.179</td>
</tr>
<tr>
<td>VILA, R.</td>
<td>I5.5</td>
</tr>
<tr>
<td>VILA, Rafael</td>
<td>I4.1, P3.184</td>
</tr>
<tr>
<td>VILBRANDT, Reinhard</td>
<td>O5C.2, P3.038</td>
</tr>
<tr>
<td>VILEMOVA, Monika</td>
<td>P3.181, P3.192</td>
</tr>
<tr>
<td>VILALOBOS, Edgardo</td>
<td>P3.014</td>
</tr>
<tr>
<td>VILLARI, R.</td>
<td>I3.3, I5.5</td>
</tr>
<tr>
<td>VILLARI, Rosaria</td>
<td>O1A.4, P1.060, P1.128, P1.129, P2.004, P2.051, P3.127, P3.131</td>
</tr>
<tr>
<td>VILLEDIEU, Eric</td>
<td>O3A.4, P1.144</td>
</tr>
<tr>
<td>VILLERS, Frantz</td>
<td>P4.157</td>
</tr>
<tr>
<td>VILLONE, Fabio</td>
<td>P2.009, P4.044</td>
</tr>
<tr>
<td>VINCENZI, Pietro</td>
<td>P2.007, P4.019, P4.020, P4.028</td>
</tr>
<tr>
<td>VINYAR, Igor</td>
<td>P4.006, P4.159</td>
</tr>
<tr>
<td>VIROT, Francois</td>
<td>O4C.4</td>
</tr>
<tr>
<td>VISCA, Eliseo</td>
<td>O4B.2, P1.107, P1.117, P1.118</td>
</tr>
<tr>
<td>VITALY, Krasilnikov</td>
<td>P2.127</td>
</tr>
<tr>
<td>VIZVARY, Zsolt</td>
<td>P3.051</td>
</tr>
<tr>
<td>VLADIMIROV, Pavel</td>
<td>P1.146</td>
</tr>
<tr>
<td>VLECEK, Jiri</td>
<td>P1.051</td>
</tr>
<tr>
<td>VOMVORIDIS, John</td>
<td>P3.025</td>
</tr>
<tr>
<td>VON DER WETH, Axel</td>
<td>O3B.2, P3.145</td>
</tr>
<tr>
<td>VON MULLER, Alexander</td>
<td>P1.104</td>
</tr>
<tr>
<td>VONDRAKEC, Petr</td>
<td>P2.059</td>
</tr>
<tr>
<td>VORA, Murtuza M.</td>
<td>P3.015</td>
</tr>
<tr>
<td>VORBRUGG, Stefan</td>
<td>O5C.3, P1.111</td>
</tr>
<tr>
<td>VORPAHL, Christian</td>
<td>P1.055, P3.054</td>
</tr>
<tr>
<td>VU, Ngoc Minh Trang</td>
<td>P2.044</td>
</tr>
<tr>
<td>VUKOLOV, Dmitry</td>
<td>P4.055, P4.056</td>
</tr>
</tbody>
</table>
VUKOLOV, Konstantin P4.050, P4.052
VULLIEZ, Karl O2C.1
VUPPUGALLA, Mahesh P4.013
VYACHESLAVOV, Leonid P3.125
W7-X TEAM P1.034, P2.068
W7-X, Team P2.122
W7-X, team P2.066
WAIBEL, Patrick P4.171
WAKAI, Eiichi P4.175
WAKATSUKI, Takuma P1.041
WALDON, Chris P2.192
WALKER, Michael P4.043
WALKER, Michael L. O5C.4
WALKER, Mike L. P4.042
WALKER, R. J. O5A.1
WALKER, Richard P3.157
WALSH, Michael P1.049, P1.055, P3.056, P3.057, P3.070, P3.203, P4.050, P4.058
WALSH, Mike P1.058
WALSH, Mike J. P4.124
WALTON, Robert P1.049, P1.055
WAMPLER, W.R. O4B.3
WAN, B.N. P1.067
WAN, Yuanxi I2.2
WANG, Bo P4.157
WANG, Dagui P2.193
WANG, Fang P3.193, P3.194
WANG, H.J. P1.067, P2.076
WANG, Haibing P4.095
WANG, Hexiang P3.039
WANG, J. P1.079
WANG, Jiaqun P3.193, P3.194
WANG, Jin P3.193, P3.194
WANG, Lijun P2.095, P2.096
WANG, Pinghuai P2.187
WANG, Qiaosen P2.095, P2.096
WANG, Shenji P1.162
WANG, Shouzhi P4.097
WANG, Shuai P1.163
WANG, Shuang P1.158, P1.159
WANG, Sonjong O1B.2, P2.031, P2.032, P4.001, P4.005
WANG, W. P4.101
WANG, Wenhao P4.097
WANG, Y. I4.3
WANG, Yingqiao P4.095
WANG, Yongbo P3.138, P4.127, P4.134
WANG, Yongfeng O4C.1
WANG, Yongsheng O2C.1
WANG, Zhijiang P1.032
WANG, Zhongwei P1.015, P1.162, P2.123
WANNER, Manfred P1.138
WARMER, F. I5.4
WARMER, Felix P4.203
WARREN, R. P4.169
WARREN, Robert P2.003
WATANABE, Kazuhiro O2C.2, P2.028
WATKINS, J.G. O4B.3
WATKINS, Jon G. P3.109
WATTS, Christopher P4.046
WAUTERS, Tom P3.034
WEBER, Thomas P3.192
WEGENER, Tobias O2A.4, O2B.2, O4A.4, P2.177
WEBNER, William O5C.4
WEI, Jing P1.187, P3.195
WEI, Shiping P1.199
<table>
<thead>
<tr>
<th>Name</th>
<th>Pages</th>
</tr>
</thead>
<tbody>
<tr>
<td>WEI, Yuguo</td>
<td>P.095</td>
</tr>
<tr>
<td>WEIHUA, Wang</td>
<td>P.068, P.193</td>
</tr>
<tr>
<td>WEIMIN, Xuan</td>
<td>P.094</td>
</tr>
<tr>
<td>WEINHORST, Bastian</td>
<td>P.026, P.060</td>
</tr>
<tr>
<td>WEINZETTL, Vladimir</td>
<td>P.057, P.058, P.3.126</td>
</tr>
<tr>
<td>WEISS, Klaus-Peter</td>
<td>P.096</td>
</tr>
<tr>
<td>WEISSGERBER, Michael</td>
<td>O5C.3</td>
</tr>
<tr>
<td>WEIBFLOG, Sven</td>
<td>P.071</td>
</tr>
<tr>
<td>WELLER, Arthur</td>
<td>P.069</td>
</tr>
<tr>
<td>WENDORF, Jorg</td>
<td>P.122</td>
</tr>
<tr>
<td>WENNINGER, Roland</td>
<td>P.020</td>
</tr>
<tr>
<td>WENZEL, Uwe</td>
<td>P.122</td>
</tr>
<tr>
<td>WERNER, Andreas</td>
<td>O5C.2, P.070, P.062, P.3.036, P.3.037, P.3.038</td>
</tr>
<tr>
<td>WERNER, Kraus</td>
<td>P.080</td>
</tr>
<tr>
<td>WESCHE, Rainer</td>
<td>O1C.4, P.089</td>
</tr>
<tr>
<td>WEST, Team</td>
<td>P.069</td>
</tr>
<tr>
<td>WEST, team</td>
<td>P.070</td>
</tr>
<tr>
<td>WHITTAKER, D.</td>
<td>O5A.1</td>
</tr>
<tr>
<td>WI, Hyungho</td>
<td>P.031</td>
</tr>
<tr>
<td>WICHTERLE, Kamil</td>
<td>P.170</td>
</tr>
<tr>
<td>WIDDOWSON, Anna</td>
<td>P.049, P.051</td>
</tr>
<tr>
<td>WIDDOWSON, Anna M</td>
<td>P.120</td>
</tr>
<tr>
<td>WILSON, David</td>
<td>P.169, P.170</td>
</tr>
<tr>
<td>WINTER, Axel</td>
<td>P.046</td>
</tr>
<tr>
<td>WIRTZ, Marius</td>
<td>O2A.3</td>
</tr>
<tr>
<td>WISCHMEIER, Marco</td>
<td>O5C.3</td>
</tr>
<tr>
<td>WOJCICK-GARGULA, A.</td>
<td>P.196</td>
</tr>
<tr>
<td>WOJCICK-GARGULA, Anna</td>
<td>P.001</td>
</tr>
<tr>
<td>WOJENSKI, Andrzej</td>
<td>P.073, P.068, P.070, P.071</td>
</tr>
<tr>
<td>WOLF, Michael J.</td>
<td>P.096</td>
</tr>
<tr>
<td>WOLF, R. C.</td>
<td>5.4</td>
</tr>
<tr>
<td>WOLFERS, Gilles</td>
<td>P.029</td>
</tr>
<tr>
<td>WOLFF, Dan</td>
<td>P.140</td>
</tr>
<tr>
<td>WOLK, Andreas</td>
<td>O5C.2</td>
</tr>
<tr>
<td>WONG, C.P.C.</td>
<td>O4B.3</td>
</tr>
<tr>
<td>WOO, Myeong Hyeon</td>
<td>P.202</td>
</tr>
<tr>
<td>WOOD, Steve</td>
<td>O3C.3</td>
</tr>
<tr>
<td>WOODWARD, Jerome</td>
<td>O3C.1</td>
</tr>
<tr>
<td>WOOLDRIDGE, Emma</td>
<td>P.036, P.037</td>
</tr>
<tr>
<td>WOONGRYOL, Lee</td>
<td>P.059</td>
</tr>
<tr>
<td>WORTH, Liam</td>
<td>P.131</td>
</tr>
<tr>
<td>WOZNICKA, Urszula</td>
<td>P.053</td>
</tr>
<tr>
<td>WU, Bin</td>
<td>P.016</td>
</tr>
<tr>
<td>WU, Chuanren</td>
<td>O5B.3</td>
</tr>
<tr>
<td>WU, Jiefeng</td>
<td>P.141, P.142</td>
</tr>
<tr>
<td>WU, Jing</td>
<td>P.143, P.127, P.4.136</td>
</tr>
<tr>
<td>WU, Xinlian</td>
<td>O2C.1</td>
</tr>
<tr>
<td>WU, Y. F.</td>
<td>P.067, P.076</td>
</tr>
<tr>
<td>WU, Yian</td>
<td>O4C.1, P.194, P.198, P.2016</td>
</tr>
<tr>
<td>WU, Yuan</td>
<td>O3B.4</td>
</tr>
<tr>
<td>WUENDERLICH, Dirk</td>
<td>P.024</td>
</tr>
<tr>
<td>WULF, Sven-Erik</td>
<td>P.182</td>
</tr>
<tr>
<td>WUNDERLICH, Dirk</td>
<td>P.025</td>
</tr>
<tr>
<td>WURDEN, Glen</td>
<td>P.122</td>
</tr>
<tr>
<td>WURSTER, Stefan</td>
<td>P.180</td>
</tr>
<tr>
<td>XI, Weibin</td>
<td>P.085</td>
</tr>
<tr>
<td>XIA, Donghui</td>
<td>P.032</td>
</tr>
<tr>
<td>XIA, Zhiwei</td>
<td>P.008, P.157</td>
</tr>
<tr>
<td>XIANFU, Yang</td>
<td>P.030</td>
</tr>
<tr>
<td>XIANMING, Zhang</td>
<td>P.030</td>
</tr>
<tr>
<td>XIANZU, Gong</td>
<td>P.068</td>
</tr>
<tr>
<td>XIAO, Bingjia</td>
<td>P.042</td>
</tr>
<tr>
<td>Author</td>
<td>Papers</td>
</tr>
<tr>
<td>-----------------</td>
<td>---------------</td>
</tr>
<tr>
<td>XIAO, Chijin</td>
<td>P2.001, P4.018</td>
</tr>
<tr>
<td>XIAO, Jixiong</td>
<td>P1.032</td>
</tr>
<tr>
<td>XIAO, Zunqi</td>
<td>P3.009</td>
</tr>
<tr>
<td>XIBILIA, Maria Gabriella</td>
<td>O3A.3</td>
</tr>
<tr>
<td>XIE, Yuanlai</td>
<td>P2.036</td>
</tr>
<tr>
<td>XIE, Zhijiang</td>
<td>P4.134</td>
</tr>
<tr>
<td>XIN, Jingping</td>
<td>P1.194</td>
</tr>
<tr>
<td>XU, Alan</td>
<td>O2A.1</td>
</tr>
<tr>
<td>XU, Hao</td>
<td>O2B.4</td>
</tr>
<tr>
<td>XU, Kun</td>
<td>P1.162</td>
</tr>
<tr>
<td>XU, M.</td>
<td>P2.076</td>
</tr>
<tr>
<td>XU, Tiejun</td>
<td>P3.195</td>
</tr>
<tr>
<td>XUAN, Weimin</td>
<td>P4.095</td>
</tr>
<tr>
<td>XUE, Yong</td>
<td>P2.002</td>
</tr>
<tr>
<td>YABUUCHI, Kiyohiro</td>
<td>P3.148</td>
</tr>
<tr>
<td>YADAV, Brijesh Kumar</td>
<td>P2.170</td>
</tr>
<tr>
<td>YADAV, Ratnakar</td>
<td>P4.013</td>
</tr>
<tr>
<td>YAGI, Juro</td>
<td>P2.172, P4.144</td>
</tr>
<tr>
<td>YAGYU, Jyunichi</td>
<td>P1.073</td>
</tr>
<tr>
<td>YAKUSIJI, Koki</td>
<td>P1.110</td>
</tr>
<tr>
<td>YAMADA, Ichihro</td>
<td>P2.075</td>
</tr>
<tr>
<td>YAMADA, Masayuki</td>
<td>P4.143</td>
</tr>
<tr>
<td>YAMADA, Tetsuya</td>
<td>P2.184</td>
</tr>
<tr>
<td>YAMAKANA, Haruhiko</td>
<td>P2.028</td>
</tr>
<tr>
<td>YAMAMOTO, Ryotaro</td>
<td>P1.206</td>
</tr>
<tr>
<td>YAMAMOTO, Ryoutarou</td>
<td>P3.167</td>
</tr>
<tr>
<td>YAMANISHI, Toshihiko</td>
<td>P4.143</td>
</tr>
<tr>
<td>YAMANOI, Kohei</td>
<td>P3.102</td>
</tr>
<tr>
<td>YAMAOKA, Nobuo</td>
<td>P4.173, P4.177, P4.187</td>
</tr>
<tr>
<td>YAMAZAKI, Masanori</td>
<td>P1.179</td>
</tr>
<tr>
<td>YAN PING, Zhao</td>
<td>P1.036</td>
</tr>
<tr>
<td>YANAGI, Nagato</td>
<td>P3.144</td>
</tr>
<tr>
<td>YANG, Jeonghun</td>
<td>P2.043</td>
</tr>
<tr>
<td>YANG, Q.</td>
<td>I4.3</td>
</tr>
<tr>
<td>YANG, Qi</td>
<td>P2.016, P3.207</td>
</tr>
<tr>
<td>YANG, Qingxi</td>
<td>O2B.4, O2C.1</td>
</tr>
<tr>
<td>YANG, Sungmoo</td>
<td>P1.080</td>
</tr>
<tr>
<td>YANG, Wanli</td>
<td>P1.160</td>
</tr>
<tr>
<td>YANG, Xinsheng</td>
<td>P4.075</td>
</tr>
<tr>
<td>YANG, Ying</td>
<td>P1.106</td>
</tr>
<tr>
<td>YANG, Yong</td>
<td>P3.076</td>
</tr>
<tr>
<td>YANG, Yu</td>
<td>P1.005, P1.008, P4.157</td>
</tr>
<tr>
<td>YANNIC, Wischet</td>
<td>P2.027</td>
</tr>
<tr>
<td>YAO, Damao</td>
<td>O3C.2, P3.195</td>
</tr>
<tr>
<td>YAO, Xinjia</td>
<td>P4.159</td>
</tr>
<tr>
<td>YASUHISA, Oya</td>
<td>P3.114</td>
</tr>
<tr>
<td>YATSUKA, Eiichi</td>
<td>P3.054</td>
</tr>
<tr>
<td>YAVORSKIJI, I.</td>
<td>P3.046</td>
</tr>
<tr>
<td>YAVORSKIJI, V.</td>
<td>P3.047, P3.048</td>
</tr>
<tr>
<td>YE, M.Y.</td>
<td>P1.067, P2.076</td>
</tr>
<tr>
<td>YE, Minyou</td>
<td>P1.015, P1.016, P1.123, P1.162</td>
</tr>
<tr>
<td>YI, Shi</td>
<td>P4.082</td>
</tr>
<tr>
<td>YIN, Dapeng</td>
<td>O2C.1</td>
</tr>
<tr>
<td>YOKOMINE, Takehiko</td>
<td>P4.187</td>
</tr>
<tr>
<td>YOLKIN, Vladimir</td>
<td>P2.106</td>
</tr>
<tr>
<td>YONEKAWA, Izuru</td>
<td>P3.023</td>
</tr>
<tr>
<td>YONG SHENG, Wang</td>
<td>P1.036</td>
</tr>
<tr>
<td>YOON, Jae Sung</td>
<td>P1.166, P3.150</td>
</tr>
<tr>
<td>YOON, Jae-Sung</td>
<td>P4.106</td>
</tr>
<tr>
<td>YOON, Siwoo</td>
<td>O1B.2</td>
</tr>
<tr>
<td>YOSHIDA, Kiyoshi</td>
<td>P2.085</td>
</tr>
<tr>
<td>YOSHIDA, Masafumi</td>
<td>O2C.2</td>
</tr>
<tr>
<td>Name</td>
<td>Pages</td>
</tr>
<tr>
<td>---------------------</td>
<td>------------------------</td>
</tr>
<tr>
<td>YOSHIHASHI, Sachiko</td>
<td>P4.173, P4.177, P4.181</td>
</tr>
<tr>
<td>YOSHIITO, Matsumura</td>
<td>P4.189</td>
</tr>
<tr>
<td>YOSHIKAWA, Satoru</td>
<td>P3.102, P3.103</td>
</tr>
<tr>
<td>YOU, Jeong Ha</td>
<td>P1.117</td>
</tr>
<tr>
<td>YOU, Jeong-Ha</td>
<td>O2A.4, O2B.1, P1.104, P1.116, P1.118, P1.119, P1.120, P2.114, P2.141, P3.131, P3.132</td>
</tr>
<tr>
<td>YU, Jie</td>
<td>O1A.3</td>
</tr>
<tr>
<td>YU, Shengpeng</td>
<td>P2.016, P3.205</td>
</tr>
<tr>
<td>YU, Sikui</td>
<td>P4.085</td>
</tr>
<tr>
<td>YU, Xiao</td>
<td>P4.085</td>
</tr>
<tr>
<td>YU, Y.</td>
<td>P1.067, P2.076</td>
</tr>
<tr>
<td>YU, Yang</td>
<td>P4.046</td>
</tr>
<tr>
<td>YUAN, B.D.</td>
<td>P1.122</td>
</tr>
<tr>
<td>YUAN, Qiping</td>
<td>P2.042</td>
</tr>
<tr>
<td>YUAN, Yinglong</td>
<td>P3.200</td>
</tr>
<tr>
<td>YUCHENG, Wu</td>
<td>P4.193</td>
</tr>
<tr>
<td>YUEFENG, Qiu</td>
<td>P4.193</td>
</tr>
<tr>
<td>YUJI, Hatano</td>
<td>P3.114</td>
</tr>
<tr>
<td>YUKI, Uemura</td>
<td>P3.114</td>
</tr>
<tr>
<td>YUN TAO, Song</td>
<td>P1.036</td>
</tr>
<tr>
<td>YUN, Sei-Hun</td>
<td>P4.153, P4.154</td>
</tr>
<tr>
<td>YUYAMA, Kenta</td>
<td>P3.173</td>
</tr>
<tr>
<td>ZABEKO, Luca</td>
<td>P1.048, P4.044, P4.046</td>
</tr>
<tr>
<td>ZABOLOTNY, Wojciech</td>
<td>P4.068, P4.070, P4.071</td>
</tr>
<tr>
<td>ZABOLOTNY, Wojciech</td>
<td>P3.073</td>
</tr>
<tr>
<td>ZACCARIA, Pierluigi</td>
<td>P1.025, P2.022, P4.008, P4.009</td>
</tr>
<tr>
<td>ZACEK, Frantisek</td>
<td>O5B.2, P2.056, P2.060</td>
</tr>
<tr>
<td>ZACHA, Pavel</td>
<td>P2.109, P2.110, P3.011, P3.119</td>
</tr>
<tr>
<td>ZACKS, Jamie</td>
<td>P2.002</td>
</tr>
<tr>
<td>ZAGORSKI, Roman</td>
<td>P1.114</td>
</tr>
<tr>
<td>ZAJAC, Jaromir</td>
<td>P2.056, P2.060</td>
</tr>
<tr>
<td>ZALEZAK, Tomas</td>
<td>P3.191</td>
</tr>
<tr>
<td>ZAMENGO, Andrea</td>
<td>O1B.3, P1.021</td>
</tr>
<tr>
<td>ZAMMUTO, I.</td>
<td>I.4.3</td>
</tr>
<tr>
<td>ZAMMUTO, Irene</td>
<td>O5C.3, P1.111, P4.123</td>
</tr>
<tr>
<td>ZAMORA, Imanol</td>
<td>P2.126, P2.204</td>
</tr>
<tr>
<td>ZAMPIVA, Enrico</td>
<td>P3.069</td>
</tr>
<tr>
<td>ZANCA, Paolo</td>
<td>O1B.3, P2.040</td>
</tr>
<tr>
<td>ZANI, Louis</td>
<td>O1C.1, P1.083, P1.088, P2.087, P2.089, P3.127</td>
</tr>
<tr>
<td>ZANINO, Roberto</td>
<td>O1C.1, P1.029, P1.084, P2.147, P3.080</td>
</tr>
<tr>
<td>ZANOTTO, Loris</td>
<td>O1B.3, P2.011, P2.023</td>
</tr>
<tr>
<td>ZAPLOTNIK, Rok</td>
<td>P4.121</td>
</tr>
<tr>
<td>ZAPRETILINA, Elena</td>
<td>P2.171</td>
</tr>
<tr>
<td>ZARAS-SZYDLOWSKA, Agnieszka</td>
<td>P2.206</td>
</tr>
<tr>
<td>ZARINS, Arturs</td>
<td>P3.170</td>
</tr>
<tr>
<td>ZASTROW, Klaus-Dieter</td>
<td>O4A.1, P2.003, P3.113, P4.061</td>
</tr>
<tr>
<td>ZAUPA, Matteo</td>
<td>P1.025</td>
</tr>
<tr>
<td>ZAURBEKOVA, Zhanna</td>
<td>P1.109, P4.116</td>
</tr>
<tr>
<td>ZAVASKIN, Janez</td>
<td>P2.175</td>
</tr>
<tr>
<td>ZEILE, Christian</td>
<td>P3.120</td>
</tr>
<tr>
<td>ZELLA, Daniele</td>
<td>P1.021</td>
</tr>
<tr>
<td>ZENG, Long</td>
<td>O2C.4</td>
</tr>
<tr>
<td>ZENG, Qin</td>
<td>P3.163, P4.084</td>
</tr>
<tr>
<td>Zhai, Xiangwei</td>
<td>P1.195</td>
</tr>
<tr>
<td>Zhai, Yulu</td>
<td>P3.058</td>
</tr>
<tr>
<td>Zhang, Bo</td>
<td>P4.157</td>
</tr>
<tr>
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