28th SOFT 2014

Symposium on Fusion Technology

September 29th / October 3rd 2014
San Sebastian, Spain

Book of Abstracts

www.soft2014.eu

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Invited speakers

Invited lecture Ref. Nr. I1.1

Progress and Planning of ITER

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The ITER Project has visibly made its transition to the construction phase during past two years. Construction activities are accelerating and the appearance of the site is changing on a daily basis minimizing any possible delay factor. All the Parties are well committed to the construction of ITER. As of mid-February 2014 commitments to In-kind procurement are approaching 89.6% in value and 70.7% (99 out of 140) in numbers of the total Procurement Arrangements. The overall construction progress is enormous. This report will focus on the progresses on buildings, the core tokamak and some of the balance of plant. With regard to building and Site Preparation, the main tokamak building, the seismic bearings were completed in early 2013 and the pouring of the "B2 slab" started at the end of 2013. The Assembly Hall, Diagnostic Building and Tritium Building slabs have been completed. The Cryostat Workshop, currently one of the largest buildings on the site has been completed. Facilities for the full site workforce have also been completed. The work on two component storage zones is underway. For Magnets, 463 tonnes of Ni3Sn, representing 96% of the ITER requirement, had been produced by December 2013. Winding equipment and associated tooling is ready in both the EU and Japan. Tooling for the PF1 coil has been procured in the Russian Federation. The Correction Coil manufacturing facility has been constructed in China. The United States winding facility for the Central Solenoid is ready. Suitable CS conductor, manufactured in Japan, has been tested at the SULTAN Facility in the EU. The Coil Feeder manufacturing facility has been completed in China. Concerning Vacuum Vessel, The manufacture of prototype segments of the vacuum vessel has either been completed or is under way in Korea. In the EU, the manufacturing design is complete and materials procurement is under way. The inner-wall shield material has been procured in India and manufacturing has started. The vacuum vessel and cryostat thermal shield material has been procured in India and manufacturing is under way in Korea. Regarding Blanket Shied and Divertor, final design approval for the Blanket Shield was achieved in 2013 and the two Procurement Arrangements signed shortly afterwards. The decision was taken in December 2013 to start ITER operation with a full tungsten divertor. As for Balance of Plant, Prototype AC/DC convertors have been manufactured in both China and Korea, whilst electro-mechanical components have been demonstrated in Russia. All the electrical distribution Procurement Arrangements have been signed with China, the EU and the United States. The IO, in close collaboration with 7 DAs, has completed the 2014 Annual Work Plan and is currently developing a realistic long-term schedule, which is expected to be submitted to the ITER Council in June 2015.

Invited lecture Ref. Nr. I1.2

Status and issues of the European contribution to ITER

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Fusion for Energy (F4E), on behalf of Europe, is responsible for the procurement of most of the high-technology items for the ITER device (e.g. parts of the superconducting toroidal and poloidal field coils, the vacuum vessel, the in-vessel components, the remote handling, the additional heating systems, the tritium plant and cryoplant and finally several diagnostic systems). In many cases the technologies required to manufacture these components are well established, in others there is still on-going design, R&D and qualification work to select and optimise the final design solutions and to consolidate the underlying technologies as, for example, in the areas of heating and current drive, plasma diagnostics, blanket first wall, test blanket modules, remote handling, etc. This invited lecture provides an update of the design and technical status of Europe's contributions to ITER and related issues. Progress in the fabrication of the double-pancake and radial plates for the toroidal field coils will be reported. The status of the qualification programme for the first wall panels will be given including the design, manufacture and testing of a reduced scale prototype. The outcome of heat testing of medium scale prototypes for the full tungsten divertor will also be reported. In relation to the heating systems, progress in the design, R&D and testing of the ion cyclotron antenna and electron cyclotron upper launched will be described as well as results from the testing of half-size ITER RF sources for the neutral beam heating using the ELISE test facility. Electromagnetic analyses of the two European test blanket modules concepts for ITER will be reported. Finally, recent developments in diagnostics systems will be described in particular for the magnetics.

Invited lecture Ref. Nr. I1.3

Experience with the comissioning of the superconducting Stellarator Wendelstein 7-X

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The stellarator Wendelstein 7-X is presently under construction at the Max-Planck-Institute for Plasma Physics in Greifswald, Germany. Assembly of the device, the periphery systems and the diagnostic and heating systems are well advanced and commissioning of the device is scheduled to start in April 2014. This is the first time since decades that a superconducting fusion device is commissioned in Europe.

The commissioning of Wendelstein 7-X consists of two major steps with increasing levels of system integration:

(I) Local commissioning of the technical components is performed with the local control system and includes instrumentation and all other peripheral systems required for the respective component. Local commissioning will be done before the end of Wendelstein 7-X assembly.

(II) Integrated commissioning is concerned with the step-wise integration of all components into one system, including the central device control and the central data acquisition system.

More in detail, the integrated commissioning consists of the following sequence:

1. Vacuum tests of the cryostat (starting April 2014)
2. Cryogenic tests of the cryostat (starting August 2014)
3. Normal-conducting coils tests (starting in September 2014)
4. Plasma vessel Vacuum tests (starting January 2015, in parallel to point 5)
5. Superconducting coils tests (including tests of the magnetic field quality)
6. Preparation for the first plasma (in April 2015)

This schedule will be followed via the detailed work packages for the device components, the peripheral systems, CoDaC (Control, Data acquisition and Communication), and the safety systems and regulations. All processes will be carried out along the guidelines of the Quality Management System, which is certified according to ISO 9001, also for the commissioning phase.

This paper briefly discusses the status of the device Wendelstein 7-X. The main body of the paper is about the planning of integrated commissioning, risk mitigation, and hands-on experiences during the important steps 1 and 2.

Invited lecture Ref. Nr. I2.1

The European Roadmap to Fusion Electricity

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With the reduction of CO_2 emissions driving future energy policy, fusion could providewith up to 30% of electricity production by 2100. This requires an ambitious, yet realistic roadmap towards the demonstration of electricity production by 2050. This talk describes *the main technical challenges on the path to fusion energy*. For all of the challenges candidate solutions have been developed and the goal of the programme is now to demonstrate that they will also work at the scale of a reactor. The roadmap has been developed within a goal-oriented approach articulated in eight different Missions. For each Mission the critical aspects for reactor application, the risks and risk mitigation strategies, the level of readiness now and after ITER and the gaps in the programme have been examined. ITER is the key facility in the roadmap and its success represents the most important overarching objectives of the EU programme. A demonstration fusion power plant (DEMO), producing net electricity for the grid at the level of a few hundreds MW is foreseen to start operation in the early 2040s. Following ITER, it will be the single step to a commercial fusion power plant. Industry must be involved early in the DEMO definition and design. The evolution of the programme requires that industry progressively shifts its role from that of provider of high-tech components to that of driver of the fusion development. Industry must be able to take full responsibility for the commercial fusion power plant after successful DEMO operation. For this reason, DEMO cannot be defined and designed by research laboratories alone, but requires the full involvement of industry in all technological and systems aspects of the design. Europe should seek all the opportunities for international collaborations. Already the Broader Approach with Japan is a good example of a positive collaboration that can give further advantages on the time scale considered here. The talk will also address the needs in the area of education and training and basic research.

Invited lecture Ref. Nr. I2.2

Engineering challenges and development of the ITER Blanket System and Divertor

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The ITER Blanket System and the Divertor are the main components which directly face the plasma. Being the first physical barrier to the plasma, they have very demanding design requirements, which include accommodating: (1) surface heat flux and neutronic volumetric heating, (2) electromagnetic loads, (3) nuclear shielding function, (4) capability of being assembled and remote-handled, (5) interfaces with other in-vessel components, and (6) high heat flux technologies and complex welded structures in the design. Following to the ITER Design Review of 2007, the main functions of the Blanket System have been substantially expanded and it has now also to provide limiting surfaces that define the plasma boundary during startup and shutdown. This has led to a redefinition of the design heat fluxes (up to 4.7 MW/m^2) and a shaping of the plasma facing surface to avoid the exposure of leading edges. This effort culminated in the Blanket Final Design Review held in April 2013 and formally closed in July 2013. The successful achievement of this milestone has allowed the Blanket System to progressively move towards the construction phase. As regards the Divertor, in September 2011, the ITER Organization proposed the possible start of the ITER operation with a full-tungsten armour Divertor in order to minimize costs and already gain operational experience with tungsten during the non-active phase of the machine. A task force was set up to fully develop the full-tungsten divertor option. The related Final Design Review was held in June 2013 and formally closed in October 2013. This has paved the ground for the ITER Council to formally decide to start the ITER operation with a full-tungsten armour Divertor in November 2013. This paper summarizes the main design and integration effort, the procurement, and the related R&D and technology qualification of the Blanket System and the Divertor.

Invited lecture Ref. Nr. I2.3

Indian Fusion Technology Programme

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To keep pace with the international advancement, the Indian national programme is being accelerated. After a mandate was given to IPR to develop magnetically confined fusion plasma experiments, the Institute indigenously developed the first Indian Tokamak, ADITYA. It was successfully commissioned in 1989 and has been generating scientific results on various topics. The next major program at the Institute for Plasma Research has been to construct a Steady State Superconducting Tokamak (SST-1) by mix of import and indigenous development. The aim of SST-1 is to 1) generate the essential database particularly for understanding the interaction between the plasma and the wall of tokamak in long pulse steady-state discharges and 2) develop various fusion relevant technologies. SST-1 has a major radius of 1.1 m and a minor radius of 0.2 m, toroidal field of 3 T and a plasma current of 220 kA. After successful engineering validation of the subsystems in integrated operations at 0.75 T to 1.5 T, successful First plasma was achieved on the 20th June, 2013. Experiments in SST-1 have started to establish the parameter regimes. It will now give us a unique opportunity to study and control the long-pulse behavior in a large aspect ratio tokamak. It will allow us to build a knowledge base for future. As a strategy towards leapfrogging to save time, IPR under DAE joined ITER as a full partner. IPR was assigned to be the nodal agency for ITER Collaboration through ITER-India as the Indian domestic agency (DA) to procure the share. In order to take full benefit from the ITER partnership complementing programs on balance/gap technology development projects have also started during the last five year plan. Preparations for developing indigenous technologies in some of these key areas of fusion have been initiated as the next step in our programme. These activities are ultimately aimed at development of our own fusion reactor for integrated testing of indigenously developed systems.Results of various experiments and technology developments will be discussed during the presentation.

Invited lecture Ref. Nr. I3.1

Programme in Support of ITER

Lorne Horton (1,2) and JET EFDA Contributors (3)

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(3) See the Appendix of F. Romanelli et al, Proc. of 24th IAEA FEC 2012, San Diego

The JET Programme in Support of ITER has completed its first phase, in which operation with the new ITER-like Wall (ILW) showed the expected reduction in retention of fuel gas and the baseline operating regimes for JET and ITER were re-established. The pattern of material erosion, transport and re-deposition has now also been (partially) quantified and found to be different in a carbon-free machine. The second phase of the exploitation of the ILW has begun with the aim of exploring the limits of operation compatible with the new wall materials. First priority in the new campaigns was a dedicated test of the effect of shallow melting of one element of JET's solid tungsten divertor. Melting was achieved in a series of seven identical discharges with very little impact on the plasma performance and no disruptions. The results of this experiment provided strong support for the ITER decision to begin operation with an all-tungsten divertor. Disruption dynamics have been found to be substantially different with the ILW so that the operation of disruption mitigation systems is now normally mandatory in JET for operation above 2.5 MA. Real-time algorithms for disruption prediction and avoidance are being used and routine disruption mitigation has been established using massive gas injection. Techniques for the generation of runaway electron beams during disruptions have been re-established and the operating space mapped. The installation of a second massive gas injection system is now complete with the programme of providing a tool for quenching or at least controlling runaway beams. The change in wall material has also affected JET operating scenarios. In addition to the expected change in divertor physics in the absence of carbon, the confined plasma transport is found to have altered. Work is underway to re-optimise these scenarios with a view to preparing for ITER operation.

Invited lecture Ref. Nr. I3.2

MAST Upgrade - Construction Status

Joseph Milnes (on behalf of the MAST-U team)

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The Mega Amp Spherical Tokamak (MAST) is the centre piece of the UK fusion research programme. In 2010, a MAST Upgrade programme was initiated with three primary objectives, to contribute to: 1) Testing reactor concepts (in particular exhaust solutions via a flexible divertor allowing Super-X and other extended leg configurations); 2) Adding to the knowledge base for ITER (by addressing important plasma physics questions and developing predictive models to help optimise ITER performance of ITER) and 3) Exploring the feasibility of using a spherical tokamak as the basis for a fusion Component Test Facility (looking at start-up, current drive, steady state behaviour, handling of high heat fluxes, plasma confinement, high beta operation and performance reliability). To deliver these objectives, a new central solenoid and TF coil components have been procured to enable plasma currents of up to 2MA, a 50% increase of TF field to 0.8T and pulses of several seconds. To control and shape the plasma, a total of 16 new coils will be fitted within the vacuum vessel and 2.5MW of NBI power will be repositioned above the machine midplane to broaden the power deposition profile. With the project mid-way through its construction phase, progress will be reported on a number of the critical subsystems. This will include assembly of the coils, armour and support structures that make up the new divertors, assembly of the new centre column coil set, installation and early commissioning of the new power supplies for powering the divertor coils and enhanced TF coil set, progress in delivering the upgraded diagnostic capability, the modification and upgrading of the NBI heating systems and the complete overhaul of the machine control infrastructure, including a new control room with full remote participation facilities. This work was funded by the RCUK Energy Programme under grant EP/I501045.

Invited lecture Ref. Nr. I3.3

DEMO diagnostics and burn control

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Reliable operation of a magnetic fusion reactor requires a robust plasma scenario combined with an integrated diagnostic and control system. Both elements together, scenario and control, have to ensure machine operation in compliance with safety requirements, achieve high plant availability in particular by keeping distance to all known operational limits, and aim for optimized fusion performance while minimizing the aging of components. Essential quantities to be measured and controlled in a tokamak reactor (and mostly also in a stellarator) are the profiles of particle densities, temperatures and plasma current, furthermore the plasma position and shape, plasma radiation, local wall loads and wall temperatures, plasma instabilities, D/T ratio and fusion power. Except for the fusion power, control schemes for most of the other quantities are already available and continuously under improvement on all current major magnetic fusion experiments. However, already for ITER and even more for a future DEMO fusion reactor, the requirements for the reliability of plasma operation are much more demanding than on any existing device. One specific problem is the stationary power exhaust, where the local power flux densities are near to design limits and must be safely controlled to avoid damage to the target plates. Regarding off-normal transient events in a tokamak reactor, the number of high-power disruptions must be minimized towards almost zero, and the few remaining disruptions have to be reliably mitigated, due to the high risk of significant damage to the first wall. While present magnetic fusion experiments are amply equipped with diagnostic and actuator systems, their implementation on DEMO will only be possible with reduced performance and/or number of systems, due to several reasons: First, the fraction of openings and voids in the breeding blanket has to be minimized in order to achieve a Tritium breeding rate TBR > 1. Second, diagnostic front end components will be subject to a harsh environment (radiation, forces, temperatures etc.) and thus may only be installed at some distance behind the first wall or blanket. Third, available actuators on fusion reactors such as magnetic field coils, auxiliary heating, gas inlets, pellet injectors and pumping systems typically can only provide slow, indirect or weak performance in DEMO. In order to achieve reliable machine operation, enhanced long-term stability of both diagnostic systems and actuators, together with redundancy in terms of both number of methods and number of channels, and finally integrated data analysis together with in-situ calibration and consistency checking methods have to be developed and implemented. Within this paper, we present a draft list of DEMO control requirements together with candidate diagnostic systems and actuators, we discuss the related implementation issues, analyse the impact of controllability on the plasma scenario definition, and summarise with an outline of necessary future R&D.

Invited lecture Ref. Nr. I4.1

IFMIF/EVEDA, the Engineering Design of IFMIF and prototypes construction under the Engineering Validation Activities

J. Knaster

IFMIF/EVEDA Project Team

IFMIF, the International Fusion Materials Irradiation Facility, will provide a suitable neutron flux to test candidate materials for the plasma facing components of future fusion reactors. Accelerator driven Li(d,xn) reactions will generate neutrons in the forward direction with a broad peak at 14 MeV and a fluence reaching >20 dpas/fpy on an available volume of 0.5 liters capable to house around 1000 small specimens. Presently in its Engineering Validation and Engineering Design Activities (EVEDA) phase under the Broader Approach (BA) Agreement between Japan Government and EURATOM, we have accomplished on schedule its Engineering Design Activities (EDA) phase [1]. In turn, its validation activities are on-going with LIPAc, a superconducting 125 mA CW deuteron Linac leading the world's linear accelerators in average beam power presently under construction in Rokkasho; ELTL, the world's largest liquid Lithium loop in Oarai; LIFUS6, the Lithium loop operational in Brasimone specifically devoted to liquid Lithium induced erosion/corrosion phenomena on RAFM steels and the High Flux Test Module prototype and Helium gas HELOKA cooling loop in Karlsruhe. The status of the validation activities has been described elsewhere [2]. Potential lessons learnt upon the stepped accomplishment of the different on-going validation activities could still be implemented in a final design towards the construction of a fusion relevant neutron source, on schedule and cost, thanks to the European-Japanese efforts within the IFMIF/EVEDA Phase.

[1] J. Knaster et al., The accomplishment of the engineering design activities of IFMIF/EVEDA the European-Japanese project towards a Li(d,xn) fusion relevant neutron source, IAEA 2014

[2] J. Knaster et al., IFMIF: overview of the validation activities, Nuclear Fusion 53 (2013) 116001

Invited lecture Ref. Nr. I4.2

Current status of the WEST Project

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This paper presents an overview of the status and relevant technical issues for the WEST Project targeted at minimizing risks for ITER divertor procurement and operation. Since the beginning of 2013, the project has entered a detailed design and manufacturing phase. The manufacturing of the divertor coils and supporting structure is in progress for delivery in September this year. Meanwhile, the divertor coils power supplies are being produced in China in collaboration with the South-Western Institute of Physics. The procurement for the first series of the divertor ITER-like target has been launched in collaboration with the European Domestic Agency. It has already revealed issues linked to the industrialization (joining processes, machining, tolerances, and testing protocols). Discussions are ongoing with the Japanese Domestic Agency for the procurement of additional divertor plasma facing units. In addition actively cooled tungsten coated plasma facing components are being developed for less loaded area (divertor baffle, upper divertor target). The design of a new steady-state ELM-resilient ICRH antenna has been completed by an international team and the manufacturing of three antennas is ongoing in China in collaboration with the institute of Plasma Physics of the Chinese Academy of Sciences. The overall diagnostic layout has been finalized and detailed studies are ongoing, keeping opportunities for further upgrade open. A new plasma control system prototyping ITER requirements is being developed in collaboration with the Institute for Plasma Research in India and European laboratories. Tore Supra internal components have now been dismantled and the assembly of the new components will start this summer. The tokamak platform is presently scheduled to be operational in early 2016.

Invited lecture Ref. Nr. I4.3

The DEMO Heating and Current Drive Programme

Presented by Minh Quang TRAN
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Under EFDA's Power Plant Physics and Technology (PPPT) Programme, a substantial amount of work regarding the Heating and Current Drive (H&CD) for a European Demonstration Fusion Power Plant (DEMO) was performed. These studies covered both physics and technology and are summarized in ref. [1]. With the creation of the EU DEMO Roadmap [2], a clear path to deliver a DEMO has been defined. In Horizon 2020, EUROfusion has then included within the PPPT framework a work package (WP) dedicated to the H&CD system for DEMO. This WP aims to provide a feasible, integrated concept design of the DEMO H&CD System, with an acceptable confidence level to meet the H&CD Requirements. These requirements are derived from the need to provide long pulse/inductive scenarios from the plasma break down to the burn phase, followed by burn control, plasma ramp down, and MHD plasma control. Wherever possible, this programme will draw upon the experience gained from the ITER R&D programme, albeit the specifications for DEMO may differ from the ITER ones. The EUROfusion H&CD WP is divided into four main technical fields of work, which are (i) electron cyclotron wave (ECW), (ii) neutral beam injection (NBI), (iii) ion cyclotron range of frequencies (ICRF) and (iv) System Engineering. The System Engineering WP is focused on capturing and organising all of the H&CD requirements, whilst ensuring that a fit for purpose H&CD system is delivered. As part of the adopted system engineering strategy, the ECW and NBI work is subdivided into conceptual design, R&D and advanced technology work streams, whereas ICRF will concentrate on conceptual antenna development and integration into the DEMO design. The development programme for ECW and NBI are intended to optimize the available technology. For the ECW system, the key developmental elements are the ECW frequency (which depends on the DEMO concept itself e.g. aspect ratio, operating scenarios and plasma profiles), the high frequency step tunable gyrotron development and the choice of the launchers required for the different tasks. In addition, flexibility with regard to the steering of the RF beam and the deposition of the power in the plasma through multiple frequency operation of the gyrotron could be gained with the introduction of an advanced broadband window. For the NBI system, including a source with minimized Cs consumption or Cs free would be a major benefit, and as such is a key developmental area. In DEMO, the efficiency of the circulating power within the plant is an important target. As such, efforts will be made to increase the efficiency of gyrotron and NBI systems through the development of advanced methods. In view of the importance of the ITER R&D for the implementation of the PPPT H&CD WP, a review of the status of the activities on ITER will be performed. A discussion on the relative cost will be attempted, taking into consideration the challenges linked to the scheme of procurement in kind foreseen for ITER. Finally the technical fields of work of the EUROfusion H&CD WP will be presented in detail.

[1]Th. Franke et al. Technological and Physics Assessments on Heating and Current Drive Systems for DEMO. This conference P 3.025
[2]F. Romanelli et al., Fusion Electricity – A roadmap to the realisation of fusion energy, EFDA, 2012

[1] G. Granucci (CNR, Milano), J. Jelonnek (KIT), M. Kalsey (CCFE), J.-M. Noterdaeme (IPP Garching), A. Simonin (CEA), P. Sonato (ENEA RFX), Th. Franke (EUROfusion PMU) and R. Verhoeven (CCFE)

Invited lecture Ref. Nr. I5.1

Status of JT-60SA construction

Pietro Barabaschi

JT-60SA Team

In 2009, after a complex start-up phase due to the necessity to carry out a re-baselining effort to fit in the original budget while aiming to retain the machine mission, performance, and experimental flexibility, the detailed design of the project was begun. In 2012, with the majority of time-critical industrial contracts in place, it was possible to establish a credible time plan, and now the project is progressing towards the first plasma in March 2019. After focused R&D and qualification tests, the procurement of the major components and plant are now well underway. In the meantime the disassembly of the JT-60U machine has been completed in 2012. The assembly of JT-60SA started in January 2013 with the installation of the cryostat base, the first item delivered from Europe, and continued in February 2014 with the installation of the three lower superconducting equilibrium field coils. Winding of the TF coils winding packs started in July 2013 in EU. A cold test facility for the TF coils has been installed at the CEA Centre in Saclay. The first TF coil will start tests in 2015. The manufacture of the EF4, EF5 and EF6 coils has been completed. All Vacuum Vessel sectors are completed and their assembly is well underway. The manufacturing of the cryostat vessel body has also begun, with final delivery planned in 2017. Contracts for all Magnet Power Supplies are placed, fabrication ongoing with the one for the Quench Protection System completed and their installation to commence in autumn 2014. On-site installation of the cryogenic system will be completed in 2015, and it will be operational late in 2016. The dual frequency 110 and 138 GHz gyrotron has made significant progress towards allowing EC heating (ECH) and current drive (ECCD) under a wide range of plasma parameters. Oscillations of 1 MW for 10 s were successful at both frequencies in a world first for a dual-frequency gyrotron by optimizing electron pitch factor using a triode electron gun. On the N-NB system, the pulse duration and the current density of the negative ion source have been successfully improved from 30 s at 80 A/m2 in the previous operation to 100 s at 120-130 A/m2. The paper will give an overview of the present status of the engineering design, manufacturing and assembly of the JT-60SA machine.

Invited lecture Ref. Nr. I5.2

Technologies for dual coolant breeding blankets

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Within the EUROfusion Power Plant Physics and Technology (PPPT) Programme (2014-2018) four breeding blanket concepts will be investigated: helium-cooled pebble bed (HCPB), helium-cooled lithium lead (HCLL), water-cooled lithium lead (WCLL) and an advanced blanket concept: the dual coolant lithium lead (DCLL). Despite the low level of maturity compared with the rest of the concepts, the DCLL has been included since this breeding blanket is one of the most promising options for tritium breeding and power extraction due to the high efficiency of its power conversion cycle. The DCLL uses self-cooled eutectic lithium lead (Li^6 enrichment 90%) as breeder, neutron multiplier and tritium carrier. Pressurized helium is used to cool the EUROFER structure, especially the first wall exposed to the plasma. Due to the self-cooled concept, the liquid metal flows at relatively high velocity to extract most of the reactor power and produced tritium. As a direct consequence of the presence of an intense magnetic field, relevant magnetohydrodynamic (MHD) phenomena are produced. Thus, flow distribution can be strongly affected causing large pressure drop, instabilities and turbulence, penalizing the blanket performance. Low conducting flow channel inserts (FCI) are used to limit magnetic (and thermal) interaction, mitigating such effects. Another important issue derived from the high PbLi velocity is the corrosion of the supporting structures in contact with the PbLi, implying the necessity of having anti-corrosion coatings. Finally, the low solubility of tritium in PbLi makes easier the extraction of the generated tritium, but it complicates the control of permeation because of the high tritium partial pressure. Permeation barriers and permeators against vacuum (PAV) -as extraction technique- are being developed in the framework of different international R+D programs to handle both challenges. Recent DCLL studies have been carried out in Europe within the TECNO_FUS Program, a Spanish national project oriented towards the development of technologies associated with breeding blankets. The project concluded with a first design of a DCLL breeding blanket, including studies on FCI fabrication, MHD and management of PbLi, among others. The present EUROfusion design follows the original DCLL concept proposed in the DEMONET study at low temperatures (maximum PbLi outlet temperatures <500°C). Major efforts are being taken to adapt the original DCLL version to new DEMO specifications, including segmentation of the box (MMS), RH requirements, new nuclear power scenario, etc. Experimental activities are planned for the upcoming five years related to FCI R&D, corrosion of EUROFER in low and high velocity PbLi flows (including development of mitigation technologies) and tritium extraction from PbLi. In this work the main results achieved during the last few years by different projects regarding DCLL (TECNO_FUS, EUROfusion…) are reviewed.

Invited lecture Ref. Nr. I5.3

Neutronics Experiments, Radiation Detectors and Nuclear Techniques. Development in the EU in Supportof the TBM Design for ITER

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The development of high quality nuclear data, radiation detectors, and instrumentation techniques for fusion technology applications is supported byFusion for Energy and conducted in a joint and collaborative effort by several European research associations. The “Consortium on Nuclear Data Studies/Experiments in Support of TBM Activities”, consisting of ENEA, KIT, JSI, NPI, AGH, and CCFE, provides related services in the experimental field aiming at the validation of nuclear data, and the prediction capabilities of neutronics simulations for ITER, IFMIF and DEMO. Essential assets of the Consortium are the availability of three 14 MeV neutron sources, a number of laboratories with key facilities (e.g. 40 MeV cyclotron, Triga reactors, strong gamma sources, tokamaks etc.) and highly-qualified scientists and engineers, able to cover a large range of experimental activities inthe field of nuclear experimentation and instrumentation, including expertise in innovative tritium measurement techniques. The experimental capabilities are complemented by high fidelity neutronic simulations, conducted on high-performance supercomputers and including sensitivity/uncertainty analyses with simulations, with the ultimate goal to test and validate the available nuclear data libraries for fusion applications. The paper first discusses the neutronics activities carried out by the Consortium, presenting : a)the past experiments related to tritium measurementsin both HCLL (Helium Cooled Lithium Lead) and HCPB (Helium Cooled Lithium Lead) mock-ups of the Test Blanket Modules designed for ITER, b)the current activities devoted to the development of neutron and tritium monitor for EU-TBM (e.g. single crystal diamond detectors, self-powered neutron detectors, neutron activation system, Mn foil activation to monitor tritium production and liquid scintillation techniques for measurement of tritium activity in LiPb etc.) and benchmark experiments for nuclear data validation. The main challenges and issues (e.g. operation in very harsh environment) are addressed as well as the proposed design solutions and available results. The current statusof the activities devoted to measure the tritium production in the TBM, using different complementary approaches, is also addressed. Another important aspect is the provision of high quality integral experimental data to support predictive fusion neutronics simulations (both at and up to 14 MeV neutron energy as well as at higher neutron energies as being relevant for the IFMIF neutron source). In this respect, a new benchmark experiment on pure copper assembly is underway devoted to test and validate the recent JEFF and FENDL neutron cross section data for fusion applications. Integral gas production crosssection measurements are also underway atneutron energies relevant to IFMIF. The unique possibility to test some of the nuclear detectors which are under investigation by the Consortium at the JET tokamak during the foreseen DT campaign in 2017/8 is also addressed and some possible experiments are discussed.

Invited lecture Ref. Nr. I6.1

Integrated european materials programme for DEMO applications: recent achievements and challenges

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The performance and reliability issues involving materials mainly for in-vessel components are foremost considerations in the successful development and deployment of future demonstration fusion reactor systems. Due to the demanding operation conditions (power conversion, longer operating periods), materials will have to sustain higher thermal, neutron and other particle loads. Therefore, many solutions that could be developed for ITER will not work in a DEMO reactor. Remarkable progress in materials development and in the increase of their maturity level has been made over the past years. Yet, there are open questions that have to be answered and knowledge gaps that have to be closed in the near future. The present paper reviews the recent progress of the EFDA Topical Group on Fusion Materials with the focus on high-heat flux materials and integrated radiation effect modeling including experimental validation. The recent progress in fusion materials development and technology is discussed. This comprises tungsten composites, self-passivation alloys, powder injection molding as well as ODS steels production. Finally, the strategy of the EUROfusion Material Project is outlined.

Invited lecture Ref. Nr. I6.2

Safety of Fusion Power Plants in View of Fission Regulations

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Since the beginning, magnetic fusion adopted safety and environmental aspects as central elements for the development of a future fusion power plant (FPP). Here, the authors summarize the major results elaborated in a study generated for a German Ministry based on previous safety reviews for different power plant concepts described in literature and assess the safety approach compared to German nuclear fission power plant (NPP) regulations. The FPP safety concept follows the NPP defence in depth principle, however, taking into account the physical and engineering differences such as the radiologic inventories, the stored internal energies, the power densities and potential release paths, necessitating other safety systems in a FPP. The major safety functions of inventory confinement, cooling and reactivity control have to be implemented by FPP and NPP, but translate into different technological solutions. For example, both plant types rely on a multiple-barrier concept regarding the confinement. The characteristics and barrier topology, however, deviate substantially. A NPP realizes barriers by each other enveloping shells while in a FPP barriers are provided by the vacuum vessel and the reactor building. With respect to cooling, FPP as NPP produce similar decay heat power levels. This aspect intrinsically demands a decay heat removal system (DHR). But, the power density in a FPP is significantly lower allowing by design measures a passive heat removal. A crucial aspect in a NPP is the reactivity control.. Nuclear power excursions are absent in a FPP limiting the reactivity control measures to plasma loss control functions ensuring the confinement integrity. In summary the study elaborates that internal events in an adequately designed FPP do not compromise confinement and thus do not lead to radiological consequences requiring off-site emergency measures. Although the radiological consequences of a FPP are significantly lower, they are still within a magnitude, for which in case of external events compromising the confinement the regulatory threshold for off-site emergency measures may be exceeded. Hence, in the future more attention has to be paid to this aspect with respect to the safety demonstration of a FPP design.

Comentario [CP1]: Eher „implemented“?

Comentario [LVB2]: Maybe this can avoid this possible misunderstanding.

Comentario [LVB3]: It is not exactly the same, namely inventory and radiotoxicity. In our publication we have discussed this distinction. Maybe: radiological consequences … are … they are

Oral presentations

Abstract Final Nr. O1A.1

Development of Fusion Fuel Cycles: Large Deviation from Defense Program Systems

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Fusion energy research is dominated by plasma physics and materials development needs while significantly less effort and funding is dedicated to fuel cycle development. The fusion fuel cycle is necessary to recycle and purify the gases required to supply fuel for the fusion engine at its required purity and throughput while meeting regulatory limits that minimize environmental impacts. With the exception of fueling for the fusion device (e.g. gas puffing or pellet injection), minimal development or funding is directed towards developing the balance of the tritium fuel cycle. One reason commonly given is that tritium technology has been developed by the defense programs of various countries, therefore further research is unnecessary and only the assembly of the fuel cycle process by engineers is needed. This paper will compare and contrast various features of the needs of fusion fuel cycle systems to the defense program tritium systems. Comparisons will be made between the source and inventory of tritium, site locations and emission requirements (including radioactive waste disposal), and tritium process requirements, i.e. batch versus continuous processing and slow versus fast processing flow rates. The paper will state the continued need to develop tritium process technologies for both defense programs and fusion fuel cycle needs.

Id 408

Abstract Final Nr. O1A.2

Finalization of the conceptual design of the auxiliary systems for the European Test Blankets

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In view of the ITER Conceptual Design Review, the design of European Test Blanket Systems has been updated and put in accordance with the ITER requirements for the present design phase. It is well known that Europe is developing two concepts of TBM, the Helium Cooled Lithium Lead (HCLL) and the Helium Cooled Pebble Bed (HCPB) one, having in common the cooling media, pressurized helium at 8MPa. TBS, namely Helium Cooling System (HCS), Coolant Purification System (CPS), Lead Lithium Loop and Tritium Extraction/Removal System (TES – TRS) have the purpose to cool down the TBM and to remove tritium to be driven to TEP from breeder and coolant. These systems are placed in PC#16, CVCS area and tritium building. Starting from the preliminary design developed in the past, detailed technical interfaces with the ITER facility have been consolidated and iterative design activities were performed to comply with design requirements/specifications requested by IO. In addition, the design of the two TBS has been consolidated with respect to Codes & Standards in view of their future licensing. The main technological open issues have been solved fixing the reference technologies for the main functions. A detailed functional analysis and the development of a preliminary DACS completed the job. In this paper the present status of design of the TBS is presented together with the preliminary integration in ITER areas.

Id 929

Abstract Final Nr. O1B.1

Reducing tritium inventory in waste from fusion devices

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This paper describes the specific challenges posed by tritiated waste from fusion machines. Two main categories of waste are concerned by tritium contamination during the whole life, operational as well as during the dismantling phase: metallic components and soft housekeeping materials (e.g. gloves, tissues, protective clothes, etc.). A part of these materials might need to be processed to reduce tritium inventory before disposal by detritiation or interim storage rather than disposed of as waste directly. A brief presentation of the reference solution for tritiated waste management in France, which is a 50-year temporary storage for tritium decay, is given. An overview of the strategic stakes provided by a treatment (mainly tritium recovery, safety management, reduction of the interim storage duration) is proposed for each radiological category of waste (VLLW, LILW-SL, ILW-LL, purely tritiated) for both metallic and soft housekeeping waste. For this latter category which is linked to the need to protect operators and to control the spread of contamination during the machine maintenance, several options of detritiation techniques by thermal treatment like heating up or incineration are described. A comparison has been made between these various technical options based on several parameters: input data in terms of facilities characteristics required for treatment, reduction of the temporary storage duration, minimisation of out-gassing rates and tritium releases into the environment, production of secondary waste, compliancy with the disposals acceptance criteria, industrial maturity of the techniques. The related issues and the preliminary results obtained are shown. The advantages and drawbacks of each process are presented from the technical but also from the safety point of view in comparison with the 50-year temporary storage.

Id 907

Abstract Final Nr. O1B.2

Nuclear analysis of Chinese fusion engineering test reactor with water-cooled ceramic breeder blanket

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China Fusion Engineering Test Reactor (CFETR) is an ITER-like superconducting tokamak reactor whose conceptual design is being conducted. Its major radius is 5.7m, minor radius is 1.6m and elongation ratio is 1.8. Its mission is to achieve 50 -200 MW of fusion power, 30% -50% of duty time factor, and tritium breeding to ensure the self-sufficiency. In order to investigate nuclear response in main components of CFETR, a detailded 3D neutronics model with 22.5 torus sector has been developed based on a close agreement of conceptual design geometry of CFETR, including water-cooled ceramic breeder blanket with heterogeneous models, shield blanket, divertor, vacuum vessel, thermal shield, toroidal and poloidal magnets, and ports. Using the Monte Carlo N-Particle Transport Code MCNP and IAEA Fusion Evaluated Nuclear Data Library FENDL, neutronics anlyses were performed. In analyses, 20 operation years are assumed corresponding to 50% of duty time. As conservative estimation, material limits as ITER (e.g. superconductor magnet and vacuum vessel) were adopted. The results of transport calculation were normalized to 200MW fusion power. In this paper, the estimation of the nuclear responses to blanket, divertor, vacuum vessel, and toroidal field coils are presented, including neutronics wall load profile, neutron flux density distribution, tritium breeding ratio profile, the nuclear heat distribution, radiation damage and gas production in material, and the radiation load to toroidal field coils taking into account possible streaming channels, i.e. gap between blanket and between shield as well as NBI duct. Some suggestions to improve design are proposed.

Id 455

Abstract Final Nr. O1C.1

Overview of Fusion Nuclear Technology and Safety Research Activities in China

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The anticipated success of ITER in controlling burning plasmas is recognized as a critical first step for the fusion energy, and R&D activities in the upcoming years must focus on the outstanding challenges faced in going to the next step, considering the high-energy neutron irradiation, high power density, complicated structures, tritium confinement, combined loads, and multiple environmental effects etc. Many of these challenges concentrate on technology development, and thorough assessment is prudent for understanding the safety aspects of the maturing technologies. To confront those challenges, China has long been performing the integrated studies on the neutron source, low activation material, thermal-hydraulics experiment and tritium technology etc. A high intensity D-T neutron source is under construction with neutron intensity ~1014n/s, aiming to verify neutronics methodology, calibrate nuclear data library and test material. The China low activation ferrite/martensitic steel (e.g. CLAM) has been developed with the irradiation test up to 20 dpa. MHD and material corrsion tests have been carried out in the liquid metal experimental loops (e.g. DRAGON-IV, LMEL-U), and transient experiments will be performed on multi-functional PbLi/He experimental loop (DRAGON-V). Tritium behaviours have been invesitigated for the PFM, structure materials and breeder. Chemistry and technology has been developed to process tritium. Moreover, several concepts of fusion test reactor are being proposed mainly to promote the fusion nuclear technology. Relying on the above study, a new project named “Key technologies for the fusion safety and radiation proctection” has been launched with the funding from MOST to support the safety assessment and regulation of the future fusion reactor in China. In this contribution, a summary of the fusion nuclear technology and safety research activities in China is presented, with the key research topics stressed in the coming years. Keywords: Fusion; Fusion technology; Fusion safety

Id 668

Abstract Final Nr. O1C.2

Fusion from the electric utilities perspective: Fusion Innovation industry Forum

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The paper presents the different future energy scenarios envisaged and the so called “Fleet Transition” in which Fusion Energy could take an important role. The foreseeing energy scenarios analysed included new developments in Energy Storage, Distributed Resource and Operational flexibility of existing electricity generation Plants. A review ofthe R&D and Innovation main drivers in the electric sector is outline, with a detail description of the main issues and strategic challenges in the medium and short term. The worldwide historical involvement of electric utilities in Fusion is presented and revised under the new USA Utilities technical assessment carried out by EPRI. The paper presented the work done in Europe in the last few years by the Fusion Industry Innovation Forum FIIF-MB in order to evaluate a wide range of fusion concepts from the utility standpoint, to enhance utilities perspective on fusion, to provide guidance to Government Bodies and National Energy strategies for fusion-utilities, and to establish a basis for communication and cooperation between the fusion community and the electric utilities. Finally the paper comments the utilities challengespointed out by the “Fusion electricity: a roadmap to the realization of fusion energy” issued this year by the EFDA (European Fusion Development Agreement).

Id 1025

Abstract Final Nr. O2A.1

Conceptual design and analysis of the helium cooled solid breeder blanket for CFETR

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To bridge the gap between ITER and DEMO and to realize the fusion energy in China, a fusion device Chinese Fusion Engineering Test Reactor (CFETR) was proposed and being designed aiming at 50–200 MW fusion power, 30–50% duty time factor, and tritium self-sustained. It was designed with two tokamak options, full superconducting tokamak and water-cooling Cu magnet tokamak, and three kinds of tritium breeding blanket concepts, including helium-cooled solid blanket, water-cooled solid blanket and liquid metal-cooled liquid blanket, have been considered for CFETR. As one of blanket candidates, a helium cooled solid breeder blanket based on the full superconducting tokamak option has been designed with emphasis on conservative design and realistic blanket technology. In this paper, the basic concept of the blanket was described and the structural design, cooling scheme as well as tritium purge scheme were introduced. It was based on use of helium-cooled low activation ferritic steel components and mixture of lithium ceramics pebbles and beryllium pebbles installed in the breeding zone. Mixture ratio of beryllium and lithium-6 enrichment were optimized for high tritium breeding ratio. Arrangements of cooling channels, breeding zones and thermohydraulic parameters of coolant were also optimized to keep the temperatures of materials in the allowable range. The main features of the blanket design were summarized together with analysis of neutronics, thermal hydraulic and mechanical performance.

Id 705

Abstract Final Nr. O2A.2

Considerations on the DEMO pellet fuelling system

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In order to harvest sufficient electrical output power, a sufficiently high operational density must be established in the Demonstration Fusion Power Reactor DEMO. Unmindful application of intense fuelling particle fluxes could cause a strong burden on both the pumping system and the fuel recovery system; most probably also causing severe problems for nuclear licensing. Hence, efficient core particle fuelling minimizing the necessary amount of fuel will be a critical issue in DEMO. First studies were performed to explore the relation between fuelling efficiency and particle deposition depth. Transport modelling investigated fuelling needs for achieving the prescribed pedestal top density of 0.85 times the Greenwald density in the European pulsed DEMO scenario. For simple gas fuelling, a flux demand of 4 x 1023 /s is estimated. Significantly less flux is required applying direct particle deposition inside the separatrix. A flux of 4 x 1022/s is calculated assuming particle deposition just inside the pedestal top. Pellet ablation and deposition modelling suggest this to be viable for pellets launched from the torus inboard. Adopting a pellet size of 6 x 1021 e-, modelling indicate the bulk of pellet particles can penetrate beyond the pedestal top already at a launch speed of 1 km/s. It appears well feasible such requirements can be fulfilled, supposed pellet technology already at hand is properly applied. Pellet systems have already proven their capability to shoot at velocities exceeding 3 km/s and launch from the torus inboard at still above 1 km/s. A study has recently been initiated in the Tritium-Fuelling-Vacuum Project of the EUROFUSION DEMO Programme working on the conceptual design of an optimized DEMO core particle fuelling system. The approach envisages an assessment of any potential matter injection technique finally aiming on design completion embedded in the DEMO fuel cycle concept.

Id 401

Abstract Final Nr. O2A.3

Post irradiation characterization of titanium beryllide grades after high temperatura irradiation up to 3000appm He production in HIDOBE-01

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Titanium beryllides are an advanced candidate material for neutron multiplier for the helium cooled pebble bed (HCPB) and/or the water cooled ceramic breeder (WCCB) breeder blankets. In the HIDOBE-01 (HIgh DOse irradiation of BEryllium) experiment , beryllium and beryllide pellets with 5at% and 7at% Ti are irradiated at four different target temperatures (Tirr): 425oC, 525oC, 650oC and 750oC up to the irradiation target of 3000 appm He production in beryllium. Pellets were supplied by JAERI (now JAEA). During post irradiation examinations the critical properties of volumetric swelling and tritium inventory were studied of the irradiated titanium beryllides and compared to the beryllium reference grade. Both titanium beryllide grades how significantly less swelling than the beryllium grade, the difference increasing with increasing target irradiation temperature. Densities of the grades were studied Archimedean immersion and by He-pycnometry, giving indications of porosity formation. While both beryllide grades show no significant reduction in density at all irradiation temperatures, the beryllium density falls steeply at higher Tirr. Finally, the tritium release and retention were studied by temperature programmed desorption (TPD). Beryllium shows the same retention trend as earlier observed in studies on beryllium pebbles, while the tritium inventory of the beryllides is significantly less, already at the lowest Tirr of 425oC.

Id 361

Abstract Final Nr. O2A.4

Permeation of hydrogen dissolved in Li-Pb under forced-convection flow through inconel tube for fusion reactor blanket loop

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In order to verify tritium recovery from fusion reactor blanket continuously, permeation of hydrogen isotopes dissolved in liquid Li-Pb is experimentally investigated under a condition of forced convection flow through an inconel-625 tube. An experimental apparatus is set up in Kyushu University to test a Li-Pb forced convection flow to simulate a tritium breeding loop. Temperature is ranged from 400oC to 600oC corresponding to blanket conditions. Hydrogen concentration disolved in Li-Pb is around 0.244 mol/m3. The Reynilds numver is 0 to 3000, and the Li-Pb flow is under a laminar to transition flow condition. The rates of overall hydrogen permeation through a dual cylindrical tube are correlated to an analytical form. The rate-determining step of hydrogen permeation is not diffusion through the inconel tube but diffusion through Li-Pb flow. The overall hydrogen permeation rates are a function of temperature, Li-Pb flow rate and initial hydrogen concentration in Li-Pb. Before starting permeation through Li-Pb flow, the hydrogen permeation from an Ar+H2 stream to the outside purge gas is meaured and the overall permeation rate is found to be under H permeation through the inconel tube. When Li-Pb flows through inside the inconel tube, hydrogen permeation rate is decreased around 1/100 lower than the intrinsic one. The rate-determining step is found to be hydrogen diffusion in Li-Pb. The overall hydrogen permeation rates are correlated in terms of dimensionless Sherwood, Reynolds and Schmidt numbers. Although the present system is a forced convection flow of a liquid metal, which has a very low Prandtle number of 0.01, its Schmidt number is estimated around 100. The boundary layer thickness is different among those for temperature, velocity and concentration. All mass-transfer data are correlated to a unique function of (Re,Sc,Sh). The results are useful for the design of a fusion reactor liquid blanket loop.

Id 251

Abstract Final Nr. O2B.1

Progress and Upgrade of KSTAR to Explore the Advanced Plasma Experiments

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The KSTAR device has been operated for 6 years since the first plasma achievement in 2008 with mission of explore the physics and technologies of high performance steady-state operation that are essential for ITER and fusion reactor. The plasma performance achieved by the 6 years operation are the reliable H-mode plasma discharge extended up to 0.9 MA in plasma current, 20 s in flattop duration, and 2.9 in normalized beta. The outstanding results of joint experiments in KSTAR are successful type-I edge localized mode (ELM) suppression by applying low-n magnetic perturbation, plasma rotation control by combined magnetic perturbation and ECH injection, and finding n=1 error field in the range of 10-5. The upgrade and experiments in KSTAR are aimed in several areas such as i) to contribute to ITER urgent research issues like ELM suppression and disruption mitigation, ii) to explore the plasma performance utilizing the KSTAR unique features of low toroidal ripple and low intrinsic error field, and iii) to prepare the scientific and engineering benchmark that are required for the K-DEMO design. The KSTAR operation campaign in 2014 will be resume from end of June with targets of 1 MA H-plasma achievement with extended ELM suppression duration over 10s and successful integrated operation of newly installed motor generator system in connection with PF superconducting magnets. In this paper, the progress of the KSTAR experiments and future plan of system upgrade will be reported in summary. The authors appreciate the strong collaboration and contribution from all the domestic and international institutions and feel thanks to our colleagues in the fusion engineering research center and KSTAR science center in NFRI. This work was supported by the Korean Ministry of Education, Science and Technology under the KSTAR project.

Id 1009

Abstract Final Nr. O2B.2

National Spherical Torus Experiment Upgrade Fabrication and Assembly

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The National Spherical Torus Experiment (NSTX) is a major U.S. facility designed to study the physics of fusion plasmas magnetically confined in a very low aspect-ratio Spherical Torus (ST) configuration. The NSTX device has been in operation since 1999, however, to further expand the operating regime and support ITER, PPPL is upgrading the device with a new, more robust center stack magnet assembly and adding a second, more tangential, neutral beam injector. The project is nearly complete with startup planning effort underway. The new center-stack magnet assembly will double the available magnetic field and plasma current while increasing the plasma pulse length from the present ~ 1 s at 0.5 T to 5 s at 1 T providing the highest performance ST facility in the world's fusion program. The NSTXU center stack contains a center core of the toroidal field (TF) magnet coils, and the ohmic heating (OH) solenoid, all enclosed in an outer Inconel vacuum liner. The design and fabrication of the new center-stack required the introduction and application of unique technological processes in order to meet rigid engineering requirements and as well as solve difficult constructability issues. Some processes PPPL employed include the use of friction stir welding (FSW), epoxy Vacuum Pressure Impregnation (VPI), new brazing and soldering processes, wire EDM, as well as others. We will describe the components of the machine and the technologies that were utilized to solve the challenging assembly and installation tasks

Id 73

Abstract Final Nr. O2B.3

Improved experiment on neutron streaming through JET Torus Hall penetrations

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Neutronics experiments are performed at JET for validating in a real fusion environment the neutronics codes and nuclear data applied in ITER nuclear analyses. In particular, the neutron fluence through the penetrations of JET torus hall is measured and compared with calculations to assess the capability of numerical tools to correctly predict the radiation streaming in the ITER biological shield penetrations. Neutron streaming experiments started in 2012 and first results were reported in [1]. Several hundreds of very sensitive thermo-luminescence detectors (TLD), enriched to different levels in 6LiF/7LiF, were used to measure the neutron and gamma dose separately. Lessons learnt from this first experiment led to significant improvements in the experimental arrangements to reduce the effects due to directional neutron source and self-shielding of TLDs. Here we report the results of measurements performed during the 2013 JET campaign. More data from new positions, further in the South labyrinth and down to the basement through the air duct chimney, were obtained. In order to avoid interference between TLDs, only TLDs containing natural Lithium and 99.97% 7Li were used. All TLDs were located in the centre of large Polyethylene (PE) moderators, with natLi and 7Li crystals evenly arranged within thin PE containers. Two PE containers were used, one in horizontal and the other in vertical orientation, to study the shadowing effect in the directional neutron field. All TLDs are calibrated in terms of neutron fluence. This improved experimental arrangement led to reduced statistical spread in the experimental data. MCNP code was used to calculate the neutron fluence at detector positions, using a JET model validated up to the magnetic limbs. JET biological shield and penetrations, the PE moderators and TLDs were modelled in detail.The measured neutron fluence was compared to the calculated fluence and C/E comparison has been obtained showing reduced agreement for positions far in the penetrations. [1] B. Obryk, P. Batistoni, S. Conroy, B. D. Syme, S. Popovichev, I. E. Stamatelatos, T. Vasilopoulou, P. Bilski, Thermo-luminescence measurements of neutron streaming through JET Torus Hall ducts, Fusion Engineering and Design, in press, doi:10.1016/j.fusengdes.2013.12.045.

Id 289

Abstract Final Nr. O2B.4

Optimization of the irradiation parameters of the DONES, alternative for the Early Neutron Source

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The DEMO Oriented Neutron Source (DONES) project is one of the alternatives presently proposed for the implementation of the Early Neutron Source (ENS) required by the DEMO design activities. It is aimed to a stepped implementation of international Fusion Materials Irradiation Facility (IFMIF) from the prototypes, presently built in the framework of the IFMIF/EVEDA project. It will consist of a D-Li stripping neutron source with an appropriate neutron spectrum to emulate fusion irradiation conditions. DONES will have the capabilities to fulfill the DEMO needs in a first step and fusion power plant needs in a second one. As a consequence of the DONES configuration it is proposed to use a single irradiation module located in the high flux irradiation area. This report shows the neutron transport calculations, using MCNP5 code, in the high flux irradiation area with the aim to evaluate in detail the sensitivity of the irradiation parameters to different configurations of the irradiation module. The irradiation parameters assessed are the He and H production (appm/s) and damage doses (dpa/s) as a function of the available volume. Besides, the effect of the other irradiation parameters like beam footprint area is also analyzed.

Id 930

Abstract Final Nr. O2C.1

Lawson Criterion of the DEMO Fusion Power Plant

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The Lawson criterion is seen today as a general energy balance equation related to plasma energy input and output. However, the goal of the Lawson criterion in its original formulation is the power balance of fusion power plants. The DEMO Fusion Power Plant (DEMO) is going to generate fusion electricity in the grid and it is required to maximize the efficiency of power plant, which can be done by increasing efficiency of individual components, especially of the blanket, turbine circuit, magnet system and heating devices. This paper establishes the Lawson criterion for the fusion power plant, based on current technological possibilities. The criterion of energy balance for the fusion power plant is created based on modern analysis using the data from available technology devices. A second equation of the energy balance is constructed for the DEMO on the basis of expected efficiencies of the scheme proposed in the construction of the DEMO. Both derived equations are compared with the original formulation of the Lawson criterion. The comparison of the equations shows that current estimates to achieve breakeven are unrealistic and the required plasma parameters at the plant must be higher. The paper also demonstrates that the assumption of achieving an ignition of the plasma is used to determine the power efficiency of the whole plant.

Id 250

Abstract Final Nr. O2C.2

The Development of Safe High Current Operation in JET-ILW

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The JET tokamak is unique amongst present fusion devices in its capability to operate at high plasma current, which provides the closest plasma parameters to ITER. The physics benefits of high current operation have to be balanced against the risks to the integrity of the machine due to high force disruptions. The installation of the ITER-Like Wall (ILW) has added risks due to the thermal fragility of the metal Plasma Facing Components (PFC) and to the observed slower timescales of the disruption process, resulting in higher disruption forces. This paper describes the operational aspects of scientific development of high current H-mode plasmas with the ILW, focussing on disruption prediction, avoidance and amelioration. In preparation for the ILW exploitation the JET real-time (RT) protection system was substantially upgraded to include an IR-based PFCs thermal monitoring system and a flexible scheduler to tailor the response to thermal alarms and other off-normal events. The RT system can, also, be configured to trigger a Massive Gas Injection, via the Disruption Mitigation Valve (DMV), to mitigate heat loads and electromagnetic forces in cases where a disruption is judged to be unavoidable. In the development of H-mode plasmas to high current, the main threats to the plasma have been addressed and different RT disruption avoidance and mitigation methods have been successfully integrated. Specific risks include plasma cooling following slow core accumulation or sudden influx of heavy impurities, harmful MHD activity, large perturbations to the plasma vertical stability or excessive heating of specific PFCs. Initial scoping studies were carried out in 2012 and achieved H-mode pulses up to 3.5 MA/3.2 T. An unintentional disruption following a large impurity influx was successfully mitigated by timely DMV triggering. The challenge for the forthcoming experimental period is the safe continuation of this programme towards a more ambitious target of 4-4.5 MA.

Id 557

Abstract Final Nr. O2C.3

A Detailed Picture of Plasma-Control System Interactions and Resonant-Like Behaviour During ELM Cycles in the Joint European Torus

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During a 2-week period in July 2012, JET produced over 120 almost identical H-mode pulses, providing two orders of magnitude more data for a single pulse type than has previously been available. A systematic statistical analysis of this data revealed the unexpected observation of a series of maxima and minima in the probability density function (pdf) for the waiting times between ELMs (arXiv:1310.0287). A search for the cause has revealed a complex interplay between the control systems and the plasma behaviour. Between ELMs the plasma cross-section calculated by EFIT moves slowly both radially outwards and vertically downwards towards the divertor, before an ELM restores its position. An oscillation with a period of approximately 8 milliseconds (125Hz frequency) is also found, that narrows the plasma and moves it downwards. The oscillations found by EFIT coincide with enhanced (or reduced) Berylium II (527nm) light emissions, and with an enhanced (or reduced) likelihood of an ELM dependent on the phase of the oscillation. ELMs tend to occur at a maximum in both the plasma's downward velocity and position, and in the rate of strike point motion across the divertor tiles. No evidence is found for the vertical control system (VCS) or shape control system being the primary cause of the observations. The plasma velocity signal used by the VCS shows no strong indication of the vertical plasma oscillations seen in EFIT reconstructions, and that evidently coincide with observations of Berylium II light emissions and ELM occurrence. The cause of the oscillations could therefore be plasma instability, or simply a small vertical oscillation that the VCS is not reacting to prevent. Whatever their cause, the observations suggest new opportunities for improved ELM mitigation and plasma control, whether by manipulating the naturally occurring oscillations to achieve ELM pacing, or for improving plasma stability.

Id 732

Abstract Final Nr. O2C.4

The enhanced pellet centrifuge launcher at ASDEX Upgrade: advanced operation and application as technology test facility for ITER and DEMO

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The pellet centrifuge at ASDEX Upgrade serves since more than 20 years as a powerful tool for plasma control. It can be used for ELM control purposes as well as for fuelling the plasma to high densities clear above the Greenwald density limit. The recently enhanced control system provides a more detailed control over parameters and an intensive view on all measured values. The cryostat system consists of two independent cryostats, thus the pellet temperature can be modified. An investigation about temperature dependence of transfer efficiency (mass loss and number of broken pellets) showed up to now a weak dependence. The knowledge about the bandwidth of suitable pellet temperatures is of interest for future continuous extrusion systems most probably used for DEMO. The main interest is on the influence of pellet temperature to the overall transfer efficiency for different geometry of transfer guiding systems. For the sake of ITER intending to establish a heating scheme with ICRF minority heating of He-3, tests are performed to inject material with pellets into the plasma using D2 –pellets as carrier. In order to test the technological possibilities to dissolve Helium in a D2 –pellet, the gas manifold was modified to provide gas mixtures in various concentrations in order to produce alloyed pellets. Admixing substances are N2 and He-4. Mixtures of H2 and D2 are investigated as well. The products were analysed with regard to extrusion temperature, mechanical properties (influence of overall efficiency) and change of concentration. A study has recently been initiated in the Tritium-Fuelling-Vacuum Project of the EUROFUSION DEMO Programme working on the conceptual design of an optimized DEMO core particle fuelling system. For this approach, first technical tests aiming on an optimized pellet transfer with respect to the preparation of the solid fuel and the transfer systems have been performed.

Id 140

Abstract Final Nr. O3A.1

Development of ITER diagnostics: neutronic analysis and radiation hardness

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The report is dedicated to problems of ITER diagnostics development caused by effects of radiation from hot plasma. An effective nuclear shielding must be arranged in a diagnostic port plug to meet the nuclear safety requirements and to provide reliable operation of the diagnostics. This task is being done with help of neutronic analysis at the design stage of integration of diagnostics within the port plugs. The task solution is demonstrated for the 11th equatorial port plug (EPP 11). The latest design models of the tokamak and the diagnostics were used. The numerical simulation includes the calculations of neutron fluxes and the gamma dose rates of irradiated materials in the port-plug, in the interspace and in the port cell. Options for nuclear shielding, such as tungsten collimators, boron carbide and water moderators, stainless steel and lead screens are examined. It was shown that it is possible to meet the safety requirements, but with some problems, mainly caused by a lack of available space in port plugs and a heavy weight of shielding components. Data for neutron fluxes along diagnostic labyrinths give us a feasibility to define radiation hardness requirements for the diagnostic components and to specify their materials properly. A question of material selecting is considered by example of glasses for windows and lenses of optical diagnostics. The results of irradiation of candidate flint and silica glasses in nuclear reactor have shown that fused silica KU-1 and KS-4V retains transparency in visible range after neutron fluence of 1E17 cm-2. Flints required for achromatic objectives have much less radiation hardness. The neutron fluence of 5E14 cm-2 produces significant degradation of flint transmittance at the wavelengths below 500 nm.

Id 573

Abstract Final Nr. O3A.2

Design and integration of lower ports for ITER diagnostic systems

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ITER diagnostic components will be installed at various positions inside the vacuum vessel and in the vacuum vessel ports at the upper, equatorial and divertor levels. This paper focuses in the design development of Diagnostic Lower Ports that will host various diagnostic systems. Some of the diagnostic components will rest in three Diagnostic Racks (Inner, Middle and Outer in order moving away from the plasma) and the port will be vacuum sealed with the Closure Plate. The Diagnostic Racks must allow the support and cooling of the diagnostics, extraction of the required diagnostic signals, and providing access and maintainability while minimizing the leakage of radiation. The Inner rack will be a cooled structure that will play an essential role for nuclear shielding since will be placed close to the divertor cassette, near the neutron source. As the divertor cassette, this structure will be cooled by water and baked at maximum temperature of 350C. Middle and Outer Racks are not water cooled and are optional, depending on diagnostic functional requirements. Each of the diagnostic racks has a maximum allowable weight of 10 tons and will be supported by the Vaccum Vessel with dedicated rails. These racks are being designed to survive the lifetime of ITER of 20 years being able to withstand all the expected nuclear, electromagnetic and seismic loads among others. The diagnostics racks are in its conceptual design phase and this paper present the current state of development including interfaces, diagnostic integration, operation and maintenance, shielding requirements, remote handling, loads cases and discussion of the main challenges coming from the severe environment and engineering requirements.

Id 696

Abstract Final Nr.	O3A.3

Engineering and Installation Challenges for the ITER Magnetic Diagnostics Flux Loops

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The Magnetic Diagnostics Flux Loops (MDFL) are a key diagnostic for the ITER tokamak, providing important information about the shape of the plasma boundary, instabilities, and magnetic error fields. In total, 237 flux loops will be installed on ITER, on the inside and outside walls of the Vacuum Vessel, and will range in area from 1m2 to 250 m2. This paper outlines the detailed engineering design of the MDFL, explaining the solutions developed to maintain measurement accuracy within their difficult operating environment: ultra-high vacuum, strong magnetic fields, high gamma and neutron radiation doses. In addition there is a requirement for very high reliability as no maintenance is possible during the 20 year machine lifetime. Installation of the system is a significant challenge and much work has been performed to optimise the design for assembly. The complexity of the in-vessel configuration means that different systems are tightly integrated, requiring development of assembly logic and tooling design. Long component delivery timescales mean component tolerances have to be dealt with by in-situ customisation. The quality of tens of thousands of components and welds needs to be ensured to minimise the as-installed failure rates, and prevent component loss. This paper discusses these challenging installation and assembly constraints, and provides significant insights for the design of essential magnetic diagnostics for ITER and beyond.

Id 198

Abstract Final Nr. O3A.4

Compendium of the experimental and design activities toward the manufacture of the in vessel viewing system for ITER

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During ITER lifetime the in vessel first wall must be inspected in order to evaluate the state of the first wall and consequently to plan maintenance activities. From this the need of providing ITER with a dedicated In-Vessel Viewing System (IVVS) in order to perform inspections and to get high resolution pictures and metrology data through which information related to damages and erosion of plasma facing components can be obtained. This is not an easy task to be implemented because inside the vessel the hardest environmental conditions are present. Integrated values of 5 MGy for gamma rays and a neutron fluence up to 2.3 1017 n/cm2 are estimated for the ITER lifetime. Furthermore ultrahigh vacuum, 120°C operative temperature and a magnetic field up to 8 T are generally present. A solution for this viewing system was already proposed and investigated by ENEA where a first prototype of the probe was designed, implemented and characterized. The paper organically describes the key concepts of the system and presents an integrated survey of the main activities and the experimental results obtained by a working team of the ENEA Frascati laboratories performed from 2009 to the end of 2013 in the framework of the GRT015 and the successive GRT282 Fusion for Energy (F4E) GRANTs. The target of the activities was to fully investigate the possible performances of the ENEA probe concept keeping into account the ITER constraints, to verify the compatibility of the key elements of the system when exposed to ITER typical working conditions and to prepare the way to the final industrialization and procurement. Several experimental activities have been performed in ENEA test facilities reproducing vacuum, temperature, neutrons and gamma irradiations and high magnetic fields conditions closely similar to the ones predicted in ITER.

Id 365

Abstract Final Nr. O3B.1

Material Erosion and Transport in JET with Metal Plasma-Facing Components: Impact on Fuel Inventory and Modification of Diagnostics Mirrors

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Since August 2011 the JET tokamak has been operated with the ITER-Like Wall (JET-ILW): beryllium (Be) in the main chamber and tungsten (W) in the divertor. i.e. the material configuration recently decided for ITER. Material erosion and fuel inventory studies are among top priorities of the JET-ILW programme. Various types of diagnostic tools, i.e. marker tiles and wall probes including test mirrors, have been employed to assess the overall material migration pattern. The specific goals of this work were to determine: (i) fuel retention in the divertor; (ii) erosion-deposition pattern of beryllium and other species; (iii) the reflectivity and surface morphology of mirrors studied within the First Mirror Test at JET for ITER. Analyses of in-vessel components have shown the erosion of Be inner wall limiters and the deposition of material on the upper tiles in the inner divertor leg. The thickest deposits, up to 15µm, contain mainly beryllium with some minority species: carbon and also nitrogen from edge cooling. Their content is low: Be/C concentration ratio >16; Be/N >45. Also fuel inventory in JET-ILW is small, both relative: Be/D >10 in deposits and absolute being below 5x1018 cm-2. This value is distinctly lower than in JET with carbon walls (JET-C) where layers of a few hundreds of micrometers were formed. The study has not identified on wall components the formation of flaking deposits which could contribute to the dust formation. It should also be stressed that the reflectivity of polycrystalline molybdenum mirrors tested on the main chamber wall was retained or even improved in some cases. This result may have a positive impact (e.g. cost reduction) on the planning and development of maintenance procedures for ITER diagnostic mirrors. The options will be presented. These findings indicate advantages of metal components in comparison to the carbon surrounding.

Id 359

Abstract Final Nr. O3B.2

Development of a high heat-flux cooling element with potential application in a near-term fusion power plant divertor

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A novel jet-impingement based cooling element has been developed with potential application in a near-term fusion power plant divertor. Combining a simplistic layered structure, together with utilisation of materials with historical pedigree within the fusion program, has minimised manufacturing complexity. The additional capacity of the design to handle high coolant pressures offers improved power cycle performance for liquids and reduced pumping powers for gases. The ANSYS Workbench software was employed to assess the performance of the design from both a thermo-fluid and static structural perspective. To verify the accuracy of the thermo-fluid work experiments were carried out on a scaled model using air as the working fluid. Heat transfer coefficient data was obtained using transient liquid crystal thermography. Numerical thermo-fluid results indicate that the design is capable of handling heat-fluxes in excess of 10 MW/m2 within the 10% pumping power limit using a number of coolants: Water, Carbon Dioxide and Helium. Heat transfer coefficients obtained from simulations were within 20% of those given by the experimental work. Static structural analysis showed that von-Mises stresses remain within the yield criterion except in highly localised areas where plastic flow would be expected. For the nuclear fusion program to be successful the development of a divertor target plate for a near-term fusion power plant is of critical importance. The cooling element developed in this work offers high heat-flux handling capability using a range of coolants together with a simplistic manufacturing process achievable with current industrial methods.

Id 605

Abstract Final Nr. O3B.3

Development of arc-discharge and plasma-sputtering methods for cleaning plasma-facing components of fusion reactors

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Accumulation of tritium in co-deposited layers on plasma-facing components (PFCs) of ITER is a potential showstopper for its operation. It is therefore important to regularly clean PFCs from deposited material without damaging their surfaces. We have tested the functionality of two different techniques, arc-discharge and plasma-sputtering methods, for cleaning ITER-relevant test samples. In the former approach, arc discharges are ignited on a sample to eject material away while in the latter case, an Ar+ plasma beam removes surface layers by sputtering. Prototype devices have been designed and constructed for the experiments and the progress of the cleaning process is monitored by a spectral detection system. In the experiments, samples with 1-um thick Al (proxy for Be) or mixed Al-W coatings on stainless steel were cleaned. In the arc-discharge device, an Ar flow of 2.0 mbar was established inside the vacuum-tight head of the cleaning system, and a 350-mm2 area was cleaned in 2 s. By scanning the head across surfaces, an A4-sized area can be treated in a few minutes. Despite the fast cleaning rates, secondary ion mass spectrometry and scanning electron microscope analyses indicate that the roughness of the samples has increased considerably (from 0.5 um to 1-5 um) and the surfaces show signs of local melting. These are, however, not critical for standard PFCs but are issues for mirror-like surfaces. The plasma-sputtering experiments were carried out in vacuum such that the distance between the Ar source and the sample was 13 cm. Coatings could be completely removed from large areas (diameter 20 cm) in 60 minutes, and no changes in surface roughness or morphology of the samples could be observed after the process. The method seems thus promising for cleaning first mirrors; the cleaning rate can be increased by using higher plasma currents and more focused Ar beams.

Id 68

Abstract Final Nr. O3B.4

Improvements in electron beam monitoring and heat flux flatness at the JUDITH 2-facility

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Three beryllium-armoured small-scale mock-ups and one semi-prototype for the ITER first wall were tested by the electron beam facility JUDITH 2 at Forschungszentrum Jülich. Both testing campaigns with cyclic loads up to 2.5 MW/m² are carried out in compliance with the extensive quality and management specifications of IO and F4E. Several dedicated calibration experiments are performed before the actual testing in order to fulfil the testing requirements and tolerances. These quality requests have been the motivation for several experimental setup improvements. The most relevant results of these activities will be presented. One major criterion is the compliance of the heat flux flatness. To verify this, a new testing set-up based on calorimetry has been developed for this particular purpose. Castellated graphite blocks with suitable geometrical dimensions, in particular with regard to the height of the single graphite columns that was predetermined via FEM analyses, are loaded by the electron gun. The loading is applied for a defined time while the temperature increase of the graphite surface is measured with a high resolution IR camera. The electron beam path pattern based on a digital scanning mode is optimized iteratively until the flatness requirements are fulfilled. Finally, having gathered important technical expertise during this optimization procedure, the heat flux variation could be decreased from ±10 % to ±5 %. A common difficulty is to detect the exact motion of the electron beam and thus the positioning of the beam path on the sample. An external triggering system forces the IR camera to make images at defined intervals. This allows the monitoring of the movement of the electron beam more easily than with the internal triggering system of the IR camera. Apart from the beam path positioning new insights have been gathered concerning the desired and actual beam dynamics, which strongly depends on the selected beam path pattern.

Id 780

Abstract Final Nr. O3C.1

Completion of the First Winding Pack for the JT-60SA TF Magnet System

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The first winding pack (WP) for the superconducting TF coils of JT-60SA tokamak has been manufactured in 2014. For their dimensions (about 7 m height and 4 m width) JT-60SA coils are, so far, the largest superconducting TF coil ever produced (only ITER TF coils will exceed them). ENEA, in the framework of Broader Approach program for the early realization of fusion with the construction of JT-60SA tokamak, awarded to ASG Superconductors in Genoa, Italy, a contract for the manufacturing of 9 TF coils of JT-60SA. This contract, signed in late 2011, implies several milestones: the manufacturing of the first WP is probably the most critical one and its achievement passed through the completion of other intermediate steps. On September 2013, indeed, the first double pancake (DP) has been produced and the remaining five DPs, that constitute the WP, have been completed by the end of 2013. WP manufacturing consists mainly of three steps: i) conductor winding; ii) DP stacking and concurrent ground insulation; iii) vacuum pressure impregnation (VPI). The choice to separate the winding process from the stacking operation, has been dictated by the conviction that any possible shape correction could be more easily achieved on a dedicated stacking station than on the winding table although at the cost of an increased number of process to be performed. The present paper describes the whole manufacturing process up to the impregnation phase, with special attention to some critical aspects encountered during winding and impregnation; indeed, due to the tight tolerances prescribed, the set-up of the winding line or the VPI process turned out to be more complex than originally expected. For instance, in the winding process one critical issue has been represented by sand-blasting performed after bending operation to prevent rollers from reducing roughness on conductor surface.

Id 884

Abstract Final Nr. O3C.2

Superconducting magnets for big tokamaks

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The development of big superconducting (SC) magnets for DEMO tokamak reactors has begun in some countries. Two such projects have been presented at the SOFT-2012. Both of them are using mainly the same design approach, which has been accepted for the ITER design. The ITER construction is going on rather satisfactorily. However ITER decisions have been chosen more than twenty years ago and now not all of them can be considered as optimal. Therefore it is reasonable to analyze some of them carefully using the experience gained under the production and operation of the SC magnets for ITER, EAST, KSTAR, SST-1, as well as LHD and W-7X projects. Certainly the use of developed and checked technologies has its advantages, but it is reasonable to reconsider some approaches to avoid the problems, which already have been met during the magnet operation: the appropriateness of forced cooled SC magnets application, current-carrying capacity degradation, displacement of thick lose twisted cable in conduit conductor (CICC) under loads, probability of breakdown on magnet feeders, winding before reaction, and some other difficulties. It is evident that the present reliability of forced cooled magnets is insufficient: they have got 18 breakdowns inside the cryostats of six forced cooled magnets (including 5 accidents in assembled devices). The high voltage tests before charging were made on all those magnets, but in good vacuum only, which is evidently not enough. The way, how to avoid the breakdowns has been suggested: the leads and feeders insulation should be as strong as that on the coils. This requirement is natural; however it is not being realized. In such a situation, it is reasonable to reconsider the bath cooling approach.

Id 356

Abstract Final Nr. O3C.3

Electrical Design Of The Inverter System BUSSARD For ASDEX Upgrade Saddle Coils

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A set of 16 in-vessel saddle coils is installed in the ASDEX Upgrade (AUG) nuclear fusion experiment for mitigation of edge localized modes (ELM) and feedback control of resistive wall modes (RWM). The coils were driven by DC current during previous campaigns, only. After two years of in-house development effort, the new inverter system “BUSSARD” has been built to the experiment. A four-phase system has been assembled to simultaneously operate up to 4 groups of coils consisting of up to 4 serial-connected coils each. The maximum current is 1.3 kA with a ripple in the range of 7 % and the frequency is variable between DC and approx. 100 Hz. The switching frequency is about 5 kHz. As a first application, rotating fields will be generated. The system can be enhanced to 16-phase operation with a bandwidth of 500 Hz. As an additional requirement, it has to be upgradable to a 24 phase system with a bandwidth of up to 3 kHz. To reach these goals, several challenges had to be solved, like: (i) damping of inverter resonances, (ii) design of proper cable input filters to avoid overvoltage at the end of up to 400 m long cables, (iii) detection of erroneous currents caused by e.g. an electrical breakdown of a vacuum isolated feedthrough, (iv) development of fast, flexible and robust controllers, (v) development of specialized interface cards to make the system inherently safe from fail operations, (vi) suppression of EMI to the experiment and the controlling electronics, (vii) DC current feeding of the inverters, (iix) appropriate connection to the AUG data network and timer signals, (ix) safety system development. An overview of the electrical design is given and discussed. First experimental results are shown as well. This project has received funding from the Euratom research and training programme 2014-2018.

Id 859

Abstract Final Nr. O3C.4

LTS and HTS High Current Conductor Development for DEMO

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The large size of the magnets for DEMO call for very large operating current in the force flow conductor. A plain extrapolation from the superconductors in use for ITER is not adequate to fulfill the technical and cost requirement. The development and prototype manufacture for high current DEMO conductors is on going at CRPP and ENEA. The baseline for DEMO magnets is a graded winding using both Nb3Sn and NbTi conductors. The toroidal field (TF) conductor has an operating current of 82 kA @ 13.6 T in the latest layout. For the high grade TF conductor, two Nb3Sn prototypes are being built in 2014 with partial support of Eurofusion. The two prototypes reflect the two approaches suggested by CRPP (react&wind method) and ENEA (wind&react method). After two yeras of design and assessment activities, industrial offers were obtained in 2013. The Nb3Sn strand (overall 200 kg) has been procured at technical specification similar to ITER. Both the Nb3Sn strand and the high RRR, cr plated copper wire (400 kg) have been delivered. The cabling trials are carried out at Tratos Cavi using equipment relevant for long length production. The completion of the manufacture of the two 20 m long prototype is expected in the second half of 2014. The performance assessment is updated on the basis of the actual results of the delivered strand. The test of the two prototypes is planned in 2015 at CRPP. In the scope of a long term technology development, outside the scope of Eurofusion, high current HTS conductors are built at CRPP and ENEA. The prototype conductor built at CRPP is designed to match the high grade conductor requirement for DEMO: it is a flat cable composed of 20 elements consisting of twisted stacks of coated conductor tape soldered into copper shells. The HTS conductor developed at ENEA in collaboration with its industrial partner TRATOS Cavi, consists of a series of REBCO 2G stacks, inserted into a slotted and twisted Al core, with a central cooling channel. Prototype samples have been manufactured in industrial environment and the scalability of the process to long production lengths has been proven.

Id 148

Abstract Final Nr. O4A.1

The European ITER Test Blanket Modules: Current status of fabrication technologies development and a way forward

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Europe has developed two reference tritium Breeder Blankets concepts that will be tested in ITER under the form of Test Blanket Modules (TBMs): i) the Helium-Cooled Lithium-Lead (HCLL) which uses the liquid Pb-16Li 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. Both concepts are using the EUROFER-97 reduced activation ferritic-martensitic (RAFM) steel as structural material and pressurized Helium technology for heat extraction (8 MPa, 300-500°C). The paper reviews fabrication technologies and procedures applied for manufacturing of the TBM sub-components, like, HCLL and HCPB cooling plates, HCLL/HCPB stiffening plates, and HCLL/HCPB first wall and side caps. The used technologies are based on fusion and diffusion welding techniques taking into account specificities of the EUROFER-97 steel. Development of a standardized procedure complying with professional codes and standards (RCC-MRx), a preliminary fabrication/welding procedure specification (pF/WPS), is described as well as a fabrication and characterization of Feasibility Mock-ups (FMU) aimed at assessing the suitability of a fabrication process for fulfilling the design and fabrication specifications. Also, fabrication procedures for the TBM box assembly are presently under development through collaboration between European Fusion Laboratories and Industry for the establishment of an optimized assembly sequence/scenario and development of standardized welding procedure specifications. Selection of optimized assembly scenario takes into account not only the design requirements and fabrication possibilities/constraints but also maximum accessibility to the welds for sound non-destructive examination (NDE) in compliance with welds classification. A future approach towards qualification of the developed fabrication technologies and procedures, through a number of medium to full-size qualification mock-ups according to European standards, is outlined before construction of the first TBMs.

Id 421

Abstract Final Nr. O4A.2

Achieving Tolerable Relative Magnetic Permeability in Austenitic Stainless Steels

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The energy confinement and stability of tokamak plasmas is significantly degraded by relatively small deviations from toroidal axi-symmetry of the confining magnetic field. Emphasis is placed on the design and installation criteria for the magnetic field generating coils, but similar care is also needed for field perturbations that occur through the magnetic response of the materials that make up the load assembly of the tokamak. While an issue common in tokamak design, the infrequent nature of build projects makes it worthwhile to formally capture this process. The MAST-Upgrade project at CCFE is delivering a highly configurable divertor through the installation of 14 new in-vessel divertor coils with associated plasma facing components and gas-sealing of the two divertors. This hardware is supported with stainless steel structures – the material being chosen for vacuum compatibility and economy. Careful control has been exercised on the delivery of components, in order to achieve sound mechanical properties together with a magnetic susceptibility low enough to reduce field perturbations to tolerable levels. This paper summarises the key actions taken by CCFE in this area. The primary method of control has been specifying an acceptable susceptibility of 5% (μr=1.05), together with heat treatment of components found to have magnetic susceptibility out of specification. A selection of heat treatment regimes of various times and temperatures has been assessed for how they reduce magnetic permeability, but also change mechanical properties and component geometry. The body of work forms a self-contained study of how to control the magnetic properties of materials during the construction of a device in which accuracy of magnetic field is significant. The study reveals the importance of both the initial forming process of raw material, and the careful selection of heat treatment regime.

Id 425

Abstract Final Nr. O4A.3

The start-up and observation of the Li target under high vacuum in the EVEDA Li test loop

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The International Fusion Materials Irradiation Facility (IFMIF) is an accelerator-based D-Li neutron source aimed at producing an intense high energy neutron flux (2 MW/m2). To realize such a condition, two 40 MeV-deuteron beams with a total current of 250 mA are injected into a liquid Li stream (the Li target) flowing at 15 m/s speed. The EVEDA (Engineering Validation and Engineering Design Activities) Lithium Test Loop (ELTL), which simulates hydraulic condition of the Li target and a purification system envisaged in the IFMIF, is a main Japanese activity of the Li target system in the IFMIF/EVEDA project. Construction and commissioning of the ELTL were completed in March 2011 in the Japan Atomic Energy Agency, and then the validation test has been performed since Sep. 2012. Following the first validation phase in which a series of tests on performance on the ELTL and preliminary validation tests on the Li target at the nominal velocity of 15 m/s were performed, the final phase of the validation activity was started in the middle of 2013. In the final phase, the stable Li target under the IFMIF condition whose Li temperature and velocity are 250 C and 15 m/s respectively under high vacuum of 10-3 Pa has been achieved so far. The procedure to start the Li target (start-up procedure) was also established to start the Li target stably. The Li target was photographed by a digital still camera and its flatness was confirmed at this time. On the other hand, measurement technics to examine stability of the Li target in detail (e.g. high speed video camera, laser-distance meter) have been developed and validated in terms of applicability to the Li target. The detailed investigation for the Li target stability will be completed by July 2014 and will be presented in the conference.

Id 372

Abstract Final Nr. O4A.4

Long-term annealing of advanced lithium metatitanate breeder pebbles

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Advanced tritium breeder pebbles, consisting of lithium-excessive lithium metatitanate, were produced by a sol-gel based fabrication and an emulsion method. In order to examine their long-term stability, the pebbles were annealed under reference purge gas atmosphere at 900 °C, which represents the temperature limit in the breeder zone of the Japanese ITER test blanket module. The samples were placed in alumina crucibles and purged with 1200 ml/h He+0.1%H2 inside a gas tight setup at a constant absolute pressure. During annealing, the oxygen and water contents were continuously monitored at the inlet and outlet of the furnace. Samples were taken after 4, 32, 64, and 128 days of annealing. The microstructure and the phase content of the pebbles were examined by scanning electron microscopy and X-ray diffractometry, respectively. The anticipated lithium metatitanate phase persists throughout the annealing without degradation. Grain growth to more than 10 µm is observed as a result of the annealing. The elemental composition of the samples was checked by inductively coupled plasma optical emission spectrometry of the main constituents. Also the open and closed porosity were measured by He-pycnometry and Hg-porosimetry. Additionally, the evolution of the pebbles' surface area was examined by multipoint BET nitrogen adsorption measurements. Crush load tests of single pebbles were used to study the changes in the mechanical behavior of the pebbles. It is observed that the mechanical strength is increased by post-sintering effects during the early stages of annealing and afterwards decreases as a result of the observed grain growth. The fabrication process does not seem to influence the evolution of the pebbles during annealing in relative terms, however, it determines the initial density and thus the absolute or initial values are set by the applied fabrication method. Ultimately both processes produce suitable pebbles for long-term use in a breeder blanket.

Id 741

Abstract Final Nr. O4B.1

Design assessment of water-cooled divertor target concepts for the European DEMO

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In the framework of the EFDA R&D program for Design Assessment Studies for the In-vessel Components, several tasks have been performed in 2013 to explore the design feasibility and heat load limits of diverse water-cooled divertor target concepts and for optimizing the design of the concepts considering DEMO-relevant operation conditions. In addition, the structural integrity of the ITER-like target model was also evaluated on the basis of failure simulation of brittle fracture and plastic fatigue. Three different divertor target concepts were considered, namely, ITER-like monoblock, modified ITER-like monoblock with a thermal break interface (both using a CuCrZr alloy tube) and tungsten monoblock with martensitic steel cooling tube. The tasks were carried out by ENEA, CCFE, CEA and IPP, respectively. A uniform reference heat flux load of 10 MW/m² was considered. For thermo-hydraulic analysis, we considered coolant bulk temperature of 200°C, velocity of 20 ms-1 and pressure of 5 MPa for the ITER-like and Thermal Break monoblocks and 325°C, 20 ms-1, 15.5 MPa for the monoblock with steel tube. Firstly, the geometry of each model (e.g. tube thickness and diameter) was optimized either by minimizing the temperature and the maximum equivalent stress or by maximising the minimum reserve factor from the SDC-IC design rules while using armour temperature, pipe temperature as constraints. For the optimized geometries, the heat load margin to the critical heat flux was estimated. Finally, static and progressive fracture mechanics simulation and cyclic plasticity analysis was carried out, respectively, for different loading and cooling conditions. Both ITER-like and Thermal Break monoblock concepts roughly satisfied the structural design criteria at 10 MW/m² whereas the critical heat load was limited to 8 MW/m² for the monoblock with steel tube. Low cycle fatigue turned out to be an issue for the Cu interlayer, but not for the Cu alloy tube.

Id 542

Abstract Final Nr. O4B.2

Development of high temperature high flux divertor for fusion power reactor

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Divertor component for conversion of high heat flux to high grade heat for fusion power reactor is developed based on some innovative technologies. Target energy flux is 20MW/m2 average, while modest transient load is also considered. Top surface of the divertor target is coated with relatively thin tungsten layer plated on a heat sink structure with under water explosive weld technique that avoids additional thermal stress during fabrication process. Heat flux is then transferred to heat sink with a composite material with enhanced thermal conductivity of 400 W/mK range. Heat sink is a closed channel of dual phase medium such as liquid metal that transfers localized and changing heat load given on the divertor target to the heat exchange section evenly and steadily as a latent heat with minimal temperature gradient. While the heat load from the plasma is 10MW/m2 level, heat is exchanged in the heat sink with a main heat transfer media, either helium or water or any other blanket coolant at the flux less than 1MW/m2. This innovative concept of divertor will provide a possible power reactor design with application of fusion energy to be converted to a high grade heat above 500 degree C while accepting harsh heat load anticipated for a near term reactor. All the key technologies described above are demonstrated experimentally including a small YAG laser that provides above 400MW/m2 level of transient heat flux simulating ELMs. Because one of the most serious feasibility issue for this concept is a transient heat load on a thin target structure, durability under a repeated high peak heat flux is being tested. This presentation will report the results of numerical and experimental results of the development of this new concept of divertor, and its implication for fusion reactor design.

Id 912

Abstract Final Nr. O4B.3

Analysis and Optimization on In-vessel Inspection Robotic System for EAST

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Since China has successfully built her first Experimental Advanced Superconducting TOKAMAK (EAST) several years ago, great interest and demand have been increasing in robotic in-vessel inspection/operation systems, by which an observation of in-vessel physical phenomenon, collection of visual information, 3D mapping and localization, even maintenance are to be possible. However, it has been raising much challenges to implement a practical and robust robotic system, due to a lot of complex constraints and expectations, e.g., high remanent working temperature (100ºC) and vacuum (10-3pa) environment even in the rest interval between plasma discharge experiments, close-up and precise inspection, operation efficiency, besides a general kinematic requirement of D shape irregular vessel. In this paper we propose an upgraded robotic system with redundant degrees of freedom (DOF) manipulator combined with a binocular vision system at the tip and a virtual reality system. A comprehensive comparison and discussion are given on the necessity and main function of the binocular vision system, path planning for inspection, fast localization, inspection efficiency and success rate in time, optimization of kinematic configuration, real-time control system, and the possibility of underactuated mechanism. A detailed design, implementation, and experiments of the end binocular vision system together with the recent development progress of the whole robotic system are reported in the later part of the paper, while, future work and expectation are described in the end.

Id 674

Abstract Final Nr. O4B.4

Integrated approach for hybrid CAD and mesh geometry based coupled multi-physics analyses

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As a recent trend, latest Monte Carlo (MC) codes such as MCNP6 are enabled to utilize unstructured meshes and Constructive Solid Geometries (CSG) hybrid geometries for neutronic calculations. This new capability offers a great flexibility for the geometry representation, and facilitates accurate heating calculations on complex geometries as required for coupled multi-physics analyses. In this work, an integrated approach has been elaborated for fully supporting the hybrid geometry MC simulation in an integrated multi-physics coupling approach. The open-source software SALOME is employed as basic platform which supplies the effective functions of CAD geometry modeling, mesh generation and data visualization. The CAD to MC geometry conversion tool McCad, developed by KIT, has been integrated into the SALOME platform and is capable of exchanging CAD data, and generating/importing unstructured meshes with the SALOME geometry and meshing modules. An important upgrade has been applied to McCad by providing support for the CAD and mesh hybrid geometry conversion for MCNP6. Furthermore, the data translation interface code McMeshTran, which was also developed by KIT and has been integrated into the SALOME platform, was extended for the processing of unstructured mesh results from MCNP6. With this improvement, neutron flux distributions on unstructured meshes can be visualized with the CAD geometry in SALOME, and the nuclear heating results can be mapped to other meshes for multi-physics coupling analyses. This approach has been verified for a test case model of the Helium Cooled Pebble Bed Test Blanket Module (HCPB TBM) of ITER. One of the breeder units in the CAD model was replaced by an unstructured mesh, and a hybrid model with CSG and mesh geometry was generated for MCNP6. The nuclear heating results on the unstructured mesh were processed and compared with the orthogonal mesh tally results calculated by MCNP5, which are showing good agreements.

Id 937

Abstract Final Nr. O4C.1

Progress of High Power and Long Pulse ECRH System in EAST

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The Electron Cyclotron Resonance Heating (ECRH) is one of the effective and attractive heating methods for the magnetic confinement fusion devices. A multi-megawatt ECH system, which would offer greater flexibility with regards to plasma shape and plasma stabilization, has been launched on Experimental Advanced Superconducting Tokamak (EAST) since 2011. The system is designed with the feature of 4 MW steerable power handling capabilities at 140 GHz, using second harmonic of the extraordinary mode(X2). The missions of the system are to provide central heating, current drive, plasma profile tailoring and control of magneto-hydrodynamic (MHD) instabilities. The ECRH system consists of 4 gyrotrons with norminal 1MW output power and 1000 sec pulse length. Each gyrotron is connected to the torus by a low-loss evacuated waveguide transmission line. The EC power can be switched either to the dummy loads for the gyrotrons conditioning or to the plasma through an in-vessel quasi-optical launcher. Considering the diverse applications of the EC system, the front steering launcher is designed to inject four independently steered beams across nearly the entire plasma cross section. The beam's launch angles can be continuously varied with the optimized scanning range of over 30° in poloidal direction and ±25° in toroidal, as well as the polarization will be adjusted during the discharge by the orientations of a pair of polarizers in the transmission line to maintain the highest absorption for different operational scenarios. This paper presents an overview of the 140GHz long pulse ECRH system, the design and the current status of the first 2MW system are also described in detailed.

Id 323

Abstract Final Nr. O4C.2

Status of the development of the EU 170GHZ /1 MW/CW Gyrotron

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In thi s paper, the statu s ofthe development of a continuous wave (CW), 1 MW, 170OHz cylindrical cavity .b'}'rotron will be presented. The scientific designs ofthe high power gyrotron component s, such as the electron gun, beam tunnel , cavity, quasi-optical output coupler and single-stage depressed col lector have been finalized in the context ofthe collaboration ofmany EU institute s'. In collaboration with the European industrial partner Thales Electron Device s (TED), the industrial design ofthe technological parts ofthe gyrotron is being completed . A CW prototype is being manufactured at TED since the beginning ofthis year, while a short pulse prototype gyrotron is under development and manufacturing at KIT since September 2013 in order to validate the design ofthe CW indu strial prototype components. The goal is to complete the first phase of RF tests on the lMW CW prototype and to validate its design and construction by end of 2015, with the short pulse gyrotron as main technological risk mitigation action. The scientific design and the progres s in the manufacturing ofthe indu strial prototype and the short pulse gyrotron will be presented. *
·n1is work was supported by Fusion for Energy under Contracts F4E-GRT-432 to the b:uropean Gyrotron Consortium (EGYC), f4E-OPE-458 to KIT and f4E-OPE-447 to TED. EGYC is a collaboration among CRPP, Switzcrland: KIT, Gcrmany; HELLAS. Grcccc; lfP-CNR, Italy. Thc vicws cxprcsscd in this publication do not ncccssaril y rcflcct thc vicws of f4E or thc Europcan Commission.

Id 745

Abstract Final Nr. O4C.3

ITER ion cyclotron H&CD system integration in ITER

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The IC H&CD system is one of the major ITER tool for achieving the plasma performances foreseen in the operation scenarios. It is designed to deliver 20MW in the frequency range 40-55MHz, during pulses up to 3600s. It is using two equatorial port plug antennas, their pre-matching and matching systems, transmission lines, Radio Frequency (RF) sources and their associated High Voltage Power Supplies (HVPS). Each subsystem includes a local controller, and a Plant System Controller (PSC) manages overall operation, safety and investment protection. A new Task Agreement has been signed by ITER Organization (IO) and F4E to manage the Antenna design work. It will have to reassess the mechanical design to satisfy the mechanical acceptance criteria and develop the design up to built to print level. Main concern is on the neutronic shielding, the tritium contamination, the different load cases to be applied on the antenna and the effect on internal component constraint. Outcomes from R&D activities will be discussed as regard to manufacturing challenges. Proposals for antenna integration have been discussed with integration team: plug gripping, guiding system, shielding plates and sealing flange brackets will be reported. The transmission lines system has reached a new step while doing the Preliminary Design Review of the gallery transmission lines. Additional input details from IO on the building penetration are required to finalize the preliminary design. Strategy to define it better will be discussed in this paper. The R&D on the RF sources is ongoing in India in order to validate the choice of the driver and final stages of the amplifier chains. Ongoing work status will be made. The design and procurement of the PSC progressed as it carried out its Preliminary Design Review in January 2014. The main outcomes and next development phase will be detailed.

Id 363

Abstract Final Nr. O4C.4

Recent progress in R&D for long pulse and ultra-high voltage components for the ITER N-NBI

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In the ITER negative-ion-based-neutral beam injector (ITER N-NBI), JAEA procures ultra-high voltage components such as 1MeV accelerator and 1MV power supply, which are required to produce a deutrium negative (D-) ion beams with 1MeV, 40A (200A/m2 in current density) for 3600s. These requirements are more than twice of those of the most powerful N-NBI of JT-60 in the world. This paper reports that R&Ds and design of these components for ITER are progressed as scheduled. In the developement of a multi-aperture, multi-grid electrostatic accelerator designed for ITER, a pulse duration and a beam energy had been limited up to 0.2s and 0. 88MeV, respectively, due to excess grid power loading by the direct interception of the negative ion beam, which is deflected by residual magntic field. The beam deflection has been well suppressed by the offset apeture of the extraction grid. This leads to the ITER-relevant beam acceleration of 0.98MeV at 185A/m2 for 0.4s. Then, the extention of the pulse duration is focused. The extraction grid is newly developed to reinforce the cooling capability. As the result, the pulse duration increases to 9s with beams of 0.88MeV, 130A/m2 without the degradation of voltage holding capability and beam current. In the development of the power supplies, the most critical component is a DC 1MV insulating transformer. The internal structure is designed to avoid the electric field concentration for long pulse duration. Instead of a conventional large porcelein bushing with 2m in a diameter, a bushing with double-layered structure is newly developed, where a small porcelein bushing is installed in a fiver reinforced plastic cylinder. This leads reduction of a diameter of the bushing to halves, consequency, reduction of cost and weight. In a full-scale mock-up test, a stable operatoion at 1.2MV with the margin of 20% was successfully achieved for 3600s.

Id 661

Abstract Final Nr. O5A.1

Comparative study of helium effects on EUROFER and EU-ODS EUROFER by nanoindentation and TEM

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It is well known that the He effects in the microstructural and mechanical properties are nowadays one of key issues in the fusion materials field. The objective of this work is to evaluate and to compare the hardness values variation with different contents of He on two reference structural materials (EUROFER and EU-ODS EUROFER) by nano-indentation and correlate it with the microstructural changes observed by TEM. Both alloys were implanted with He ions in a stair-like profile configuration using energies from 2 MeV (~ 750 appm) to 15 MeV (~ 350 appm) at room temperature. Nanoindentation study at room temperature has been carried out on as-received state (normalized + tempered) and on implanted state. The nanoindentation tests on He implanted samples showed a hardness increase that depends on the He concentration. The maximum hardness increase observed was 41% in EUROFER and 21% in EU-ODS EUROFER and corresponded with the zone with the highest He concentration which is around 750 appm at 5 mN [1]. This behaviour is mainly attributed to the Y203 particles in the EU-ODS EUROFER steel, which could act as defects sink. FIB lamellas of each steel were extracted with two different implanted He concentrations (750 appm and 350 appm). TEM investigations carried out in the lamella with maximum He content has shown He nano-bubbles formation on both alloys. These microstructural features seem to be the cause of the hardness increase measured by nanoindentation. [1] Roldán, M.; Fernández, P.; Rams, J.; Jiménez-Rey, D.; Ortiz, C. J.; Vila, R., Journal of Nuclear Materials 2014, 448 (1–3), 301-309.

Id 723

Abstract Final Nr. O5A.2

The RF RAFMS RUSFER-EK-181 – Physical Properties

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The RF reduce activation ferritic-martensitic steel (RAFMS) RUSFER-EK-181 (Fe-12Cr-2W-V-Ta-B) as the radiation and heat resistance engineering reference material for cores of fusion and fission (fast) nuclear reactors are under manufacture. Industrial heats and products are obtained. The paper will give the physical properties of the RUSFER-EK-181 concerning: - Fabrication, Production and Heat treatment regimes. - Chemical composition, Phase states, Critical points (martensitic, magnetic, Ac1/Ac3). - Radiation damage, Nuclear transmutations and Cooling after long time neutron irradiation in fusion (DEMO-RF) and fission (BN-600, BOR-60, IVV-2M) nuclear reactors used to test fusion reactor materials. Comparability between fission and fusion irradiation effects is evaluated on the basis of calculations of damage correlation parameters like transmutation, displacements per atom (dpa) and gas production (H/dpa, He/dpa). - Specific heat (-50 C – 1000 C); Diffusivity (-50 C – 1000 C); Thermal expansion (RT – 1000 C); Electrical resistivity (-150 C – 1000 C); Thermal conductivity (-50 C – 1000 C); Magnetic characteristics (magnetization, coercive force, RT); Density (-50 C – 1000 C); Elastic module (RT – 750 C). The properties given apply to the reference steel with the specified alloying and impurity ranges and with the specified thermo-mechanical heat treatment. Some properties (thermal conductivity, heat capacity, thermal expansion, diffusivity, elastic module, magnetic) are relatively independent of the exact composition and the heat treatment regimes. These can be used directly. Other properties (radiation damage, transmutation, cooling, gas production) can be determined by scaling and interpolation of results from bounding compositions. Such scaling and interpolation needs to be justified by showing consistent and continuous trends in material behavior with compositional variation. However, it is understood that when such properties are measured directly for the reference steel, these direct measurements will replace the ones determined from compositional scaling and interpolation. Understanding these properties is essential for a reliable prediction of in-service behavior of steel.

Id 891

Abstract Final Nr. O5A.3

The SITE-SICF/SIC composite: fabrication and properties

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Continuous SiC-fiber-reinforced SiC composite (SiCf/SiC) is an attractive candidate structural material for advanced concepts of future fusion power plants, mainly due to the favourable intrinsic properties of the SiC ceramics, i.e. high temperature- and chemical stability, low neutron activation and afterheat levels as well as due to the fact that it is the only non-magnetic material proposed. Fabrication of such composites is a very challenging task due to limitations and requirements set for fusion-relevant structural materials. The SITE, a recently introduced route for fabrication of 3D-SiCf/SiC composites, and the properties of the prepared material will be presented. By using the process, which involves filling the 3D woven SiC-fibre preform with a mixture of submicron and nanosized SiC powder as a passive filler and further infiltration by pre-ceramic polymer and heat treatment, a composite with low and fine porosity was achieved. Homogeneous microstructure, high matrix crystalinity and favourable grain size contributed to relatively high thermal conductivity (~ 60 W/mK at room temperature and 30 W/mK at 1000 °C) in comparison to other state-of-art materials. Further room for improvement was identified by heating the SiC-based composite to coarsen the microstructure. The main remaining issue for the material is its poor strength, which results from the inactive (too thin and in some places non-existent) PyC interphase layer.

Id 1016

Abstract Final Nr. O5A.4

Fabrication of TBMs cooling structures demonstrators using additive manufacturing (AM) technology and AM+HIP

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Currently, the European ITER program is strongly working on two concepts of TBMs (Test Blanket Modules) known as HCLL (Helium-Cooled Led- Lithium) and HCPB (Helium-Cooled Pebble Bed). TBMs are composed of several subcomponents. One of them are cooling plates, that exhibit high structural complexity, intricate geometries and embedded cooling channels with very thin walls which are subjected to extreme conditions during operation: He coolant pressure of 8 MPa and temperature up to 550 °C. Such complexity makes very difficult to build cooling plates following conventional manufacturing process, like diffusion welding via HIP, because high thickness accuracy (±5%) and micro-joinings are required In this sense, AM technologies emerge as a promising alternative to produce complex TBM subcomponents. In this work several demonstrators, each of them consisting of six rectangular channels with dimensions according to those specified for the EU TBMs, were manufactured using ferritic-martensitic 9%Cr-1%Mo-V steel by selective laser manufacturing (SLM) technology. This steel has a metallurgical behaviour similar to EUROFER steel, the reference structural material for DEMO blanket concepts due to its reduced activation properties under severe irradiation conditions. Optimal SLM parameters led to an as-built density of 99.35% and a final density of 99.74% after hot isostatic pressing (HIP). Dimensional control shows that differences between the original design and the component are not higher than 100 microns The microstructure of the demonstrators was investigated using optical and scanning electron microscopy prior and after HIP treatment. In both cases, the microstructure consists in very thin martensitic laths. Mini tensile samples were extracted from the demonstrators by wire EDM to measure the mechanical tensile properties before and after HIP. Hardness was measured before and after HIP treatment. The feasibility of manufacturing complex TBM designs made of ferritic-martensitic steels by SLM was fully demonstrated as world premiere.

Id 989

Abstract Final Nr. O5B.1

Overview of the ITER Hot Cell Complex

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A new design of the ITER Hot Cell Complex is proposed in this paper. The ITER Hot Cell Complex consists of the Hot Cell Building, the Radwaste Building and the Personnel Access Control Building. These three buildings support the ITER Tokamak Complex during operation and maintenance periods and are here proposed to be modified from earlier versions to improve safety and to account for new or previously unaccounted for Project requirements. The Hot Cell Building houses the following main functions, the remote maintenance of in-vessel components, the maintenance of remote handling equipment that is used inside the Tokamak to maintain the in-vessel components) and the processing of medium level waste (Type B) coming from the Tokamak. The building is a substantial concrete nuclear building on five levels (two below grade) that maintains confinement using a centralized HVAC and Detritiation System. The Radwaste Building houses the processing of the low level waste (Type A). This can be either liquid or solid. The building is a concrete nuclear building on three levels (one below grade) that maintains confinement with a once-through HVAC system. The Personnel Access Control Building is a three-storey steel-framed building that has many functions, including, the control of personnel access to all of the ITER nuclear buildings, the hosting of a number of radiological and environmental laboratories, the very low level waste treatment process (TFA), control rooms for Remote Handling and Radwaste and a Bunker, used for the Back-Up Control Room. This paper describes the integration of the facilities within these three buildings, explaining their function and the trade-offs made in the design. In general, the new design of the Hot Cell Complex either maintains the safety envelope of the previous design or improves it.

Id 743

Abstract Final Nr. O5B.2

Use of Virtual Reality for optimizing the life cycle of a Fusion component

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Efficient development of a complex system such as a fusion component needs a stringent integration of standard and new constraints. For example, compared to the previous fusion experimental devices, remote handling (RH) and safety requirements are in ITER key parameters which must be integrated since the earliest design. For optimizing such integration studies, CEA, IRFM decided in 2010 to implement the use of virtual reality (VR) tools during the life cycle (from design to operation) of a fusion component. This paper describes a first feedback of such use for fusion engineering purposes. After a short overview of the CEA, IRFM VR platform capabilities, three main uses will be described: design review, simulation of remote handling and hands-on operations, with man in the loop. Design review mode was intensively used within the framework of a fruitful collaboration with ITER design integration team. This mode, fully compatible with CAD software, enables scale one data visualisation with stereoscopic rendering. It improves the efficiency in detecting inconsistencies inside models and machine sub-system design optimisation needs. Several accessibility cases of major Safety Important Components (SIC-1) were studied giving important requirements to the design at an early stage. CEA, IRFM, in close collaboration with expertise of CEA, LIST for VR simulation software, applies VR technologies for designing RH maintenance scenarii for ITER Test Blanket Module and ICH&CD Port Plugs. RH compatibility studies using VR pointed out major design drivers while helping to propose credible solution. VR platform is intensively used in the design of WEST components and assembly studies, providing important information about the feasibility of assembly processes, optimisation of physical mock-ups and ergonomic posture and gestures of operator. Finally, new perspectives, as the integration of safety constraints (dose calculation) will be described, demonstrating the powerful of VR tools at different stages of the component lifecycle.

Id 599

Abstract Final Nr. O5B.3

Development of Laser Beam Welding for the Lip Seal Configuration

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A vacuum seal using the lip sealing technique is emerging as the most likely choice for fusion devices, to comply with the requirement of maintainability. The welding technology considered for Lip sealing is laser welding, due to the attributes of small spot diameter, low concentrated heat input and high precision. While the process is established for thin plates, it is important to carry out a study of lip seal on application specific configuration, to assess the dependence of beam parameter like, laser power, speed and focusing distance on penetration and quality of weld joint. Further, the assessment of the effect of welding set-up variables like air-gap between plates, plate misalignment, and laser beam misalignment on the weld quality is also required. An additional issue is the availability of very limited information for penetration requirements and for examinations to check the weld integrity in codes and standards (e.g. From RCC-MR/ASME). An experiment has therefore been conducted to study the effect of various variables on samples size of 150mm*50mm having 2mm thickness. To retain the requirement of weld depth, a similar configuration of Tube-to-Tubesheet joint of Heat exchanger is taken where penetration requirement is ≥ Wall thickness, with a target penetration of ~3mm. The results indicate that by using a 3.3kWCO2 CW laser with a speed of 3m/min and focal length of 250mm, the penetration of 3mm can be achieved with Heat affected zone of ~2mm. Destructive, Non-destructive and He-leak examination has been carried out to ascertain the tightness of weld. Experimental recommendations are air gap should be <150microns; Beam misalignment and plate misalignment should be <200microns and 100microns resp. The paper presents the results of this study and also the plan for developing a large (3.0mx8.0m) size laser welded seal, that simulates, appropriately, the configuration required in large dimension fusion devices.

Id 379

Abstract Final Nr. O5B.4

Vacuum Tight Threaded Junctions (VTTJ): a new solution for reliable heterogeneous junctions in ITER

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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 (high temperature and high mechanical loads). 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, showing excellent leak tightness in vacuum conditions and high mechanical resistance. The main advantages with respect to existing technologies (for example, friction welding and electron beam welding) 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 is proposed for ITER, in particular for the ITER Heat and Current Drive Neutral Beam Injector and for its prototype to be tested at Consorzio RFX, the MITICA experiment. This paper gives a detailed description of the VTTJ technique, of the samples manufactured and of the qualification tests that have been carried out so far.

Id 428

Abstract Final Nr. O5C.1

Optimisation of a Lower Hybrid Current Drive launcher for ITER

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An international R&D program for LHCD in ITER is being conducted to deliver 20MW (CW) with N||peak=2.0±0.2 for different scenarios. The launcher is based on the passive-active mulitjunction (PAM), a concept that, as was already confirmed experimentally, is more resilient to density conditions in the shadow of the port, notably ELMs; offers good coupling under strong power reflection from plasmas, particularly near the cut-off density (and with a vacuum gap); has good directivity; and allows long-pulse operation. A road map was established to accomplish three major goals: optimise the overall performance of the PAM (power balance, phasing, N|| spectrum, coupling and directivity); minimise the E-field enhancement in its secondary waveguides (resulting from the recycling effect of multijunctions) that, via RF breakdown, may hamper its power handling capability; and shorten the launcher, to reduce weight and increase the space allocated for neutron shielding. The objective is to achieve this concurrently, by exploring the geometrical parameters of the PAM (the output waveguide widths, to decrease Emax at the launcher mouth; the phase shifter height, to shorten the PAM; and, more importantly, the bijunction lengths, to optimise the performance and minimise Emax in the secondary waveguides), investigating an individual 4A/5P PAM module in detail, before assessing a full row (6 modules; 24A/25P guides), with, additionally, all parameters studied in the density range from cut-off (nec) to 10×nec (expected during ELMs) using good resolution in both density and multijunction length. Besides the “2010 design” (the reference for ITER), this study also addresses an alternative configuration having the same waveguide structure, but a different phase-shifter arrangement, more straightforward to optimise and that could reduce Emax inside the launcher, therefore making it attractive for ITER. An overview of the main results will be presented, and the pros and cons of each design compared and discussed.

Id 533

Abstract Final Nr. O5C.2

Status of the R&D activities to the design of an ITER core CXRS diagnostic system

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The CXRS (Charge-eXchange Recombination Spectroscopy) diagnostic for the core plasma of ITER will be designed to provide observation of the dedicated diagnostic beam (DNB) over a wide radial range, roughly from r =0.7 to close to the plasma axis. The collected light will be transported through the Upper Port Plug #3 (UPP3) to a bundle of fibres and ultimately to a set of remote spectrometers. The design is particularly challenging in view of the ITER environment of particle, heat and neutron fluxes, temperature cycles, electromagnetic loads, vibrations, expected material degradation and fatigue, constraints against tritium penetration, integration in the plug and limited opportunities for maintenance. Moreover, a high performance is expected (étendue × transmission, dynamic range) since the beam attenuation is large and the background light omnipresent, especially in terms of bremsstrahlung, line radiation and reflections. The first mirror, which is facing the plasma, is obviously the most vulnerable component. It is subjected to simultaneous erosion and deposition of impurities which may reduce the specular reflectivity significantly. A shutter is foreseen to protect it from the particle fluxes when no measurement is taking place. The present contribution will give an overview of the current status and activities which deal with the core CXRS system, summarizing the investigations which have taken place before entering the actual development and design phase. Physics studies were performed in relation to atomic physics and data evaluation, as well as erosion and deposition on the first mirror. Engineering studies were conducted on several concepts and on specific components like shutter, retractable tubes, mirrors and mirror mounts, calibration systems, etc. They cover the optical design, electromagnetic, structural and thermal analysis of single components and assemblies and lead to preliminary performance assessments. Details are given in numerous separate, specialised contributions.

Id 630

Abstract Final Nr. O5C.3

Data acquisition remote node powered over the communications optical fiber

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Large nuclear fusion reactors, like ITER, will have harsh and noisy electromagnetic environments nearby the machine. Foreseeing the necessity for special data acquisition remote nodes, on difficult access locations and as close as possible to the experimental devices, has driving the design of the present work. The ongoing IPFN development of such system is based on the power-over-fiber technology recent advancements. System implementation aim is to attain a proof of concept for the fusion technology field and others e.g. particle physics, industry, etc. The design allows the replacement of noise susceptible copper cables, from diagnostics, by the communications optical fiber of the data acquisition remote node. Optical fibers provide galvanic isolation, are inherently immune to high electromagnetic noisy environments and simultaneously can supply power to the remote node electronics, accomplishing autonomy for power peaks consumption. System architecture uses a laser power converter (array of photovoltaic cells) to convert the laser light from the optical fiber into electricity. The electrical energy is stored on a super-capacitor for powering the remote node electronic components, such as an ADC, a small FPGA and an optical transmitter. The laser power converter is also used for the communications receiver implementation of the remote node. A description of the so far system implementation and obtained results is presented.

Id 277

Abstract Final Nr. O5C.4

Technology readiness level assessment of neutron diagnostics for DEMO

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Neutron diagnostics will play an increasing role in plasma control, plasma science and safety in DEMO. There is limited space for diagnostics in order to maximise tritium breeding ratio and the harsh radiation environment will significantly degrade the performance of many traditional diagnostics. Neutron diagnostics can be used both for safety analysis such as tritium production rate calculations or a variety of plasma parameters such as total neutron yield and hence fusion power. Recently diagnostics have improved such that complex plasma phenomenon such as MHD instabilities can also be observed using neutron diagnostics. This work assigns a Technology Readiness Level for different neutron diagnostic systems compared to a DEMO ready system based on a review of current developments in neutron diagnostic systems. Technology readiness levels (TRLs) are a recognised method of evaluating technology development programs and reducing the risk. In this assessment the TRL scale is focused on testing in increasingly relevant environments, ranging from lab scale testing, to 14MeV generators through to JET, ITER and finally DEMO, each step representing an increase in neutron flux and environmental challenges. The assessment shows that diagnostic systems based on fission chambers have the highest TRL and represent a low risk method of measuring fusion power. Neutron diagnostic systems for tritium production rate measurement are in development but so far have only been tested at 14MeV generators. The assessment clearly shows the importance of ITER for testing fusion neutron diagnostics and improving their readiness level. Several gaps are identified which need to be addressed in future work programmes in order to accelerate neutron diagnostic development for DEMO. This work was funded by the RCUK Energy Programme under grant EP/I501045 and the European Communities under the contract of Association between EURATOM and CCFE and conducted partly under EFDA PPPT (WP13-DAS04).

Id 986

Posters

Abstract Final Nr. P1.001

Status of design study for CFETR Tokamak machine

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The design and research work of China Fusion Engineering Test Reactor (CFETR) tokamak machine started in ASIPP since from 2012, the main concept of engineering design and relevant key technology for CFETR machine have come to end. The key R & D works will be the next step. A huge number of design changes and optimizations of CFETR machine have been launched with the purpose of getting high operation parameters and available maintenance method. The magnet system (TF PS CS &CC magnets) has been detailed designed according to physical parameters, such as major raidus R=5.7 m, 160 VS volt seconds provided by CS coils, 5.0 T magnetic field in the palsma center and so on. The main safety assessment work of superconducting magnet has been discussed smoothly. Besides, the progress of in-vessel key components such as vacuum vessel, thermal shielding, tritium blanket and divertor are present in this paper. The duty time will be 30%~50%. The structure design work and neutronic analysis of blanket system have been completed. The TBR can meet the requirement of the design of CFETR. In addition,in this paper the considerations of the maintenance scheme of the Blanket and Divertor are presented. The key issues such as the operational conditions, functions, mechanical design of the Remote Handling Maintenance System (RHMS) and the maintenance time estimations for high plant availability are also introduced with the purpose of safty assessment.

Id 246

Abstract Final Nr. P1.002

The development of multi-lithium ball injecting system driven by high pressure gas for disruption mitigation on EAST

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Disruption mitigation system is one of the indispensable subsystems of ITER. killer pellet and massive gas injection are two promising methods for disruption mitigation, but these two technology and methods have many deficiencies, so it is significance to developing a new technology for ITER. The aim of this project is to develop a multi-lithium ball hybrid injection system by using high pressure gas for disruption mitigation. This system contain two parts, the first part is a fast valve which is driven by eddy current, and it could be opened within 0.3ms. The second part is a automatic supply system of multi-lithium ball, which could provide accurately number of lithium ball that needed. The new technology has a lot of advantages on disrution mitigation, first of all ,the high pressure gas and lithium ball could be injecied into the plasma at the same time ,and the high pressue gas could get into the boundary area of plasma , the lithium ball could get into the center area of plasma ,so it could cool down the temperature of core and boundary area plasma at the same time, which is very important for disruption mitigation. Secondly, lithium is a light metal, its mass number is 7, by injecting lithiu ball , the thermal radiation will become more stronger. Thirdly, lithium is a promising coating material on EAST, the application of lithium couldn’t bring bad influence for next discharge. EAST is the first full superconducting tokamak in the world, which has similar structure with ITER, Developing a new technology for disruption mitigation on EAST is a very meaningful thing not only to EAST but to ITER, it could provied a effective tool for disruption mitigation on EAST, and the research results also could give a good reference to ITER.

Id 285

Abstract Final Nr. P1.003

Mock-ups Qualification and Prototypes Manufacture for ITER Current Leads

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Three types of high temperature superconducting (HTS) current leads are designed to carry 68kA, 55kA or 10kA to the ITER magnets. Before the supply of the HTS current lead series, the design and manufacturing process is qualified through mock-ups and prototypes. Seven mock-ups, representing the critical technologies of the current leads were built and tested successfully in the Institute of Plasma Physics of the Chinese Academy of Sciences (ASIPP) in 2013. After the qualification some design features of the HTS leads were updated. This paper summarizes the qualification through mock-ups. The lessons learned are also discussed. In 2014 ASIPP will start the manufacture of the prototypes. The preparation and manufacturing process are also described. The material and joints were qualified. Residual resistance ratios (RRR) of 80% C10100 samples were higher than 300. The performance of the heat exchanger (HEX) with RRR 200 was analyzed and compared with RRR 100. The temperature deviation was less than 10%. The brazing joint, soldering joint and EB welding joint were qualified after the leak test, tensile test or joint resistance test. HEX, low temperature superconducting (LTS) linker, EB welding and Instrumentation mock-ups were tested. The pressure drop ~22kPa for the HEX of toroidal field (TF) lead and ~6kPa for the HEX of correction coil (CC) leads were achieved. Less than 5% voids in the soldered joint of the CC LTS linker was found. The destructive test on the twin box joint was also applied. No visible defects were found. The critical currents of the HTS stacks were tested to be larger than 400A (CC) and 800A (TF) in liquid nitrogen. The soldering of stacks to the shunt was qualified. Many tools such as the soldering, nickel removing, sub-cable shaping, assembly, shipping as well for the production were updated.

Id 451

Abstract Final Nr. P1.004

Manufacture and test of the CTB&SBB thermal shield prototype for the ITER Magnet Feeder system

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The system of International Thermonuclear Experimental Reactor (ITER) feeders brings power, helium cooling and instrumentation to the magnets, terminated by S-bend and coil terminal boxes (CTB&SBB) outside the cryostat. An 80K rectangle AL thermal shield is hung inside CTB&SBB to reduce the thermal radiation heat loads. Al tubes have cross-section of square (22 mm 22 mm) shape at outside to have good contact with flat panels and a circular hole (diameter of 18 mm) at inside for helium flow. ASIPP will supply all 31 sets feeders for ITER. A CTB&SBB thermal shield prototype is manufactured and tested for qualification before the series production. The detailed configuration of this rectangle AL thermal shield has been presented in this paper firstly. Besides, the more points of manufacturing process of this thermal shield, especially the welding process and procedure for ensuring the good welding quality and less than total 5mm deformation on such 7.3m long thermal shield during welding with designed special tooling are introduced in this paper. In addition, the cold test and results, including the implement of cooling down with 13bar and 17.5g/s 80K He gas, the temperature distribution on different panels of thermal shield are also presented. This whole process of manufacture and test lay a good foundation for the series production.

Id 450

Abstract Final Nr. P1.005

Study on the welding process of the CTB out box prototype of ITER

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CTB(Coil Terminal Box) out box is a very important component of ITER(International Thermonuclear Experimental Reactor) feeder system. It supply a vacuum environment about 10-3Pa for the thermal shield and all internal components. The box size is about 8000×1200×1500mm, welded by 40mm thick stainless steel(304L) plate. There are many interfaces with high position precision on the box for various valves, overpressure port, vacuum barrier and some other special components. So the control of the welding deformation is the key processing technology of the box, and the four nearly 8 meters long weld lines are the main factors affecting the precision of the box. In this paper, the main welding process of the box have been studied. The narrow gap MIG(Metal Inert-Gas welding) was use for the box by automatic welding machine. Firstly, a simple simulation of the welding process for the four long weld lines on CTB out box was carried out by the use of the finite element analysis software SYSWELD. The results of deformation and residual stress distribution gave references for the process improvement and welding tooling design. Then a 2m length mock up box was welded for qualifying the welding parameter and deformation distribution. The tooling of the prototype box was designed for control the deformation during the welding process. After the main welding of the CTB outbox prototype was finished, the dimension and nondestructive inspections results show that the welding deformation of the 8m length side plate can be less than 3mm, and the quality of all the weld seams can meet the requirements of ITER.

Id 578

Abstract Final Nr. P1.006

Design, Manufacture and Commissioning of the Final Test Cryostat System for the ITER Central Solenoid Modules

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The ITER Central Solenoid system is composed of 6 individual Central Solenoid Modules (CSMs) that are assembled together at the ITER site to form a 12.8 m high superconducting magnet. General Atomics (GA) has been awarded a contract by UT-Battelle c/o Oak Ridge National Laboratory (ORNL) for the fabrication of the CSMs. The Final Test Cryostat System (FTCS) is used in conjunction with other subsystems to perform final testing of the CSMs before shipping. Final testing includes vacuum leak testing, Paschen testing, cryogenic testing and full current superconducting tests. The latter tests will generate high magnetic fields in and around the FTCS. The FTCS includes the test chamber, test chamber hood lift system, 80K thermal shield, cold mass support structure, connecting bridge and load path, vacuum pumping system, helium leak detector and Paschen test system and has been subcontracted from GA to Babcock Noell (BNG) incl. training for Paschen testing. The major steps in design of the FTCS are described and a summary of the manufacturing state is given.

Id 143

Abstract Final Nr. P1.007

Shielding Optimisation of the ITER ICH&CD Antenna for Shutdown Dose Rate

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The Ion Cyclotron Resonance Heating (ICRH) antenna is designed to couple RF heating and current drive into the ITER plasma, and will reside in equatorial port plugs 13 and 15. Shutdown dose rates (SDR) within the port interspace are required to be less than 100 µSv/hr after 106 seconds cooling, in regions where maintenance access is required. Previous analysis has demonstrated the adequacy of the antenna internal shielding; however a significant contribution to the SDR results from neutrons streaming down the gaps between the port frame and vessel extension, which in common with other ITER port plugs leads to increased activation of surrounding structures. The mitigation of this streaming is the main subject of the presented analyses. An updated MCNP model of the antenna was created to reflect the latest design, which was integrated into a variant of the B-lite ITER reference model with modified equatorial blankets. Steel shielding plates in the port gaps were proposed to attenuate streaming neutrons, and scoping studies were conducted to assess the effectiveness of several configurations. A configuration was then selected (a front dogleg arrangement), and subjected to high resolution 3-D activation analysis using MCR2S for mesh-coupled transport-activation calculations. It was concluded that the selected configuration was able to reduce the SDR from ~500 µSv/h to 220 µSv/h (still in excess of dose requirements). Approximately 30% of this was due to neighbouring ports; in isolation the ICRH port was shown to result in an SDR of 160 µSv/h. Beams of increased dose rate were observed in the port interspace along the lines of sight of the removable vacuum transmission lines. An angular biasing scheme was implemented into the MCR2S decay gamma source routine to resolve this effect, concluding that the beams were typically 300–500 µSv/h, which may require further design considerations. This work was funded by F4E under contract F4E-2008-OPE-002-01-07.

Id 261

Abstract Final Nr. P1.008

Operational Experience of the JET Neutral Beam Actively Cooled Duct Liner and Implications for ITER Operations

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Neutral beam injection systems have proved themselves as the most effective form of auxiliary heating in tokamak plasmas. A fundamental limitation on pulse length has been the effect of re-ionised neutral particles in the restricted drift space or ‘duct’ between the beamline and tokamak. These re-ionised particles are typically focused onto the duct wall by the tokamak magnetic field which produces significant power density on the duct wall. In the case of JET these power densities can be of the order of 10MW/m2 which lead to significant temperature rises, evolution of gas trapped in the wall surface, further re-ionisation and thus further duct wall heating. To extend the beam pulse length in JET an actively cooled duct wall liner was installed in 2011 as part of the EP2 upgrade which would limit temperature rises in the duct wall. The intention was to achieve a steady state equilibrium between duct pressure and duct wall temperature, allowing up to 20 second beam pulses in JET and demonstrate the feasibility of significantly longer beam pulses in future multi-megawatt beamlines such as those for ITER. This paper describes the operational experience of the JET actively cooled duct liner and compares it with the previous inertially cooled duct. The paper discusses the initial conditioning of the duct liner during restart commissioning and subsequent high power performance. The new duct was conditioned in 48 plasma pulses during its first time commissioning in only 35 plasma pulses subsequently. The inertial duct liner conditioning typically required 100 plasma pulses. In a well conditioned beamline for a 15MW beam pulse the duct liner temperature and pressure were observed to establish an equilibrium 5 seconds into the pulse and remain in a steady state to the end of the pulse. The observed behaviour is compared with the design basis predictions.

Id 601

Abstract Final Nr. P1.009

A new Disruption Mitigation System for deuterium – tritium operation at JET

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Disruptions, the fast accidental losses of the plasma current and stored energy in Tokamaks, are a critical issue for reactor •]scale fusion facilities like ITER and present a risk of severe damage to vital plant components and structures. At JET, the Tokamak experiment closest to ITER in terms of operating parameters and size, Massive Gas Injection (MGI) has been established as an active machine protection method. As a • glast resort • h measure it reduces heat fluxes and mitigates halo currents during disruptionswhich potentially have a serious impact on the beryllium and tungsten plasma •]facing materials of the main chamber and divertor. For the upcoming deuterium •]tritium (DT) experiments at JET, a new Disruption Mitigation System (DMS) has been designed and installed to provide: 1. Good reliability of the used equipment due to the restricted plant access during DT operation. 2. Utilisation of technologies compatible with DT operation and restriction of common and industrially used components and materials (e.g.integrated circuits, PTFE etc.) to minimise the risk of damage by radiation and neutrons. In addition this new DMS enables MGI studies with two mitigation systems at different locations and geometries during deuterium operation. This article presents the new DMS at JET, which consists of an all metal special gate valve compatible to high pressure injections, a fast high pressure eddy current driven valve, a high voltage power supply and an all metal gas handling system providing six supply lines for flammable and noble gases. The fast valve is located on a horizontal port (inner diameter 150 mm) and is 1.6m away from the vacuum vessel wall. The valve throughput varies with the injection pressure (efficiency 80 •]98%); the maximum injected amount of gas is approximately 4.0 • ~10 •]3 MPa.m3 (at maximum system pressure of 5.0 MPa).

Id 758

Abstract Final Nr. P1.010

Radiation levels in the ITER Tokamak Complex during and after Plasma Operation

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Extensive neutronics and 3-D activation simulations were carried out to assess the levels of radiation throughout the ITER tokamak complex, which comprises the tokamak, diagnostics and tritium handling buildings. The simulated radiation sources included D-T fusion neutrons exiting the cryostat and gamma rays arising from the activation of cooling water and activated pipe chases. Resultant biological dose rates, dose rates to silicon and particle fluxes, for both neutrons and gamma rays, have been calculated and are mapped throughout the complex. Radiation fields during plasma operation due to activated water show photon biological dose rates approaching 3200 Sv/hr in close proximity to the upper cooling pipes, whilst dose rates on the B2 level of the tokamak complex exceed 0.1 μSv/hr inside the diagnostics and tritium handling buildings. Neutrons originating from inside the tokamak have been shown to result in dose rates of less than a Sv/hr inside the port cells (during plasma operation). The dose rate from activated steel pipe chases 106 seconds after shutdown were on the order of 1 μSv/hr. Novel simulation techniques were applied to assess radiation fields in high fidelity throughout the tokamak complex during cask transfer movements of activated divertor and blanket modules, for a three weeks post operation maintenance scenario. Simulations of integrated dose to electronics based on a ‘smeared source’ show that for multiple divertor cask transfers (54 cassette transfers in total), the maximum integrated dose to silicon is 84.5 Gy, observed inside the port cell. The dose inside the tritium handling building is shown to be of the order of 1x10-7 Gy. The simulation data have been approved by ITER and consolidated into the ITER Radiation Maps tool in order to plan maintenance activities. Further studies will be needed to update the radiation maps datasets as the design of the building is finalised. This work was funded by the ITER Organisation under contract ITER/CT/13/4300000734.

Id 945

Abstract Final Nr. P1.011

On the study of catalytic membrane reactor for water detritiation: modeling approach

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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 involved in these systems, tritium bonded in tritiated carbonaceous species or water can be recovered 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. Hence, models used to describe mass transports in membranes reactors are based on the hypothesis that thermodynamic equilibriums of isotope exchange reactions are reached. In a previous study, a sensitivity analysis done on the main operating parameters of a catalytic membrane reactor used for heavy water dedeuteration has shown that thermodynamic equilibriums of isotope exchange reaction may not be systematically reached [1]. In this paper, the influence of reaction kinetics is evaluated thanks to a 2-D steady-state heterogeneous model coupling mass, thermal and momentum transfers. Perturbations of gas phase species distribution due to catalyst particles presence are accounted for by effective properties (dispersions in the gas bulk and Knudsen diffusion in catalyst pores). This allows to link macroscopic entity (e.g. inlet/outlet flow rate or partial pressure) to species production/consumption rates at the catalyst's active site scale. Finally, the results of the sensitivity analysis done with this model were compared to experimental data using D2O as a simulant of tritiated water. This led to the simplification of the reaction scheme by identification of rate determining steps. The phenomenological model proposed is a then a good starting point for future scale-up studies.

[1] Mascarade J., Liger K., Joulia X., Troulay M., Meyer X-M., Perrais C.: "On the study of catalytic membrane reactor for water detritiation: a sensitivity analysis on operating parameters", 10th International Conference on Tritium Science and Technology, 21-25 October 2013, Nice (France)

Id 532

Abstract Final Nr. P1.012

Status of the Cold Test Facility for the JT-60SA Tokamak Toroidal Field Coils

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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 will be provided by European industry and tested in a Cold Test Facility (CTF) at CEA Saclay. The coils will be tested at the nominal current of 25.7 kA and will be cooled with supercritical helium between 5 K and 7.5 K to check the temperature margin against a quench. The main objective of these tests is to check the TF coils performance and hence mitigate the fabrication risks. The most important components of the facility are: its 11.5 x 6.5 meters large cryostat in which the TF coils will be installed under vacuum; the 500 W helium refrigerator and the valve box to cool the coils down to 5 K and circulate 24 g/s of supercritical helium through the winding pack made of a cable in conduit conductor; the power supply and the HTS current leads to energize the coil; the control and instrumentation equipment (sensors, PLC's, supervision system, fast data acquisition system, etc.) and the Magnet Safety System (MSS) that protects the coils in case of quench. The paper will give an overview of the design of this large facility and give the status of its realization.

Id 273

Abstract Final Nr. P1.013

Construction of a Test Platform for Test Blanket Module (TBM) Systems integration and maintenance in ITER Port Cell #16

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The testing of Tritium Breeder Blanket concepts is one of the ITER missions and has been recognized as an essential milestone in the development of a future reactor ensuring tritium self-sufficiency, extraction of high grade heat and electricity production. The Test Blanket Module Program in ITER has been established in 2008. Europe is currently developing two reference breeder blankets concepts that will be tested in ITER under the form of Test Blanket Modules (TBMs): • the Helium-Cooled Lithium-Lead (HCLL) concept, which uses the eutectic Pb-16Li as both breeder and neutron multiplier, and • the Helium-Cooled Pebble-Bed (HCPB) concept, which features lithiated ceramic pebbles (Li4SiO4 or Li2TiO3) as breeder and beryllium pebbles as neutron multiplier. In the frame of the SUSEN project, funded jointly by the EU and the Czech Ministry of Education, investments up to 95 M€ in the areas of fission and fusion are envisaged. The goal of SUSEN is to build up a modern research infrastructure and experimental facilities for research programmes in these areas. All investments will be executed by mid-2015 and the new infrastructure will be fully available from beginning of 2016. In the area of fusion investments within SUSEN, a part of a new experimental building will be dedicated to the construction and installation of a test platform. The experimental TBM platform will simulate the Port Cell #16 hosting the European HCLL and HCPB TBMs, as well as the mock-ups of TBM systems, structures and main interfaces in view of carrying up tests and validations of integration/maintenance operations for the European TBM systems. This paper describes the test platform, foreseen to be built in the Czech Republic, and the maintenance procedures to be tested on the facility, such as pipes cutting, pipes welding, equipment removal etc.

Id 736

Abstract Final Nr. P1.014

3D Printed fusion components concepts and validation

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The geometric complexity and high accuracy simultaneously required in magnetic fusion devices, particularly stellarators and tokamaks, hampers the production of fusion components and devices. Rapid manufacturing construction methods, particularly enhanced for fusion, may contribute to a faster cycle and lower cost production of certain components for tokamaks and stellarators. Casting, cutting, forming, welding and mechanising are conventional production techniques for major fusion components, i.e. coil casings, coil frames, vacuum vessels and blankets. Synergies may emerge by combination of additive manufacturing (3D printing) with conventional manufacturing methods. 3D printing combined with resin moulding is tested by construction of the coil frame and the vacuum vessel of a small stellarator, UST_2. Satisfactory coil frames have been obtained by moulding acrylic resin in a special 3D printed polyamide hollow three-dimensional structure. The conceptual engineering design, construction process and validation of the components are described. The presented manufacturing method might contribute to advance the future 3D printing of larger metallic components for fusion.

Id 406

Abstract Final Nr. P1.016

Status of the CNESM diagnostic for SPIDER

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The ITER neutral beam test facility under construction in Padova will host two experimental devices: SPIDER, a 100 kV negative H/D RF source, and MITICA, a full scale, 1MeV deuterium beam injector. A detection system called Close-contact Neutron Emission Surface Mapping (CNESM) is under development with the aim to resolve the horizontal beam intensity profile in MITICA and one of the eight beamlet groups in SPIDER, with a spatial resolution of 3 and 5 cm2 respectively. This is achieved by the evaluation of the map of the neutron emission due to interaction of the deuterium beam with the deuterons implanted in the beam dump surface. CNESM uses nGEM detectors, i.e. GEM detectors equipped with a cathode that also serves as neutron-proton converter foil. The diagnostic will be placed right behind the SPIDER and MITICA beam dump, i.e. in an UHV environment, but the nGEM detectors need to operate at atmospheric pressure, so to contain the detector a vacuum sealed box has been designed to be installed inside the vacuum vessel and at atmospheric pressure inside. The box design was driven by the need to minimize the neutron attenuation and the distance between the beam dump surface and the detector active area. This paper presents the status of the CNSESM diagnostics and describes the design of the detectors and of the sealed box (in particular the analysis carried out to define its parameters, the necessary pumping and leak test procedures to ensure the compatibility of the box with the UHV environment and the proposed installation/removal procedure). Also the general layout of the diagnostic as part of the SPIDER experiment will be discussed. Finally the preliminary design of MITICA CNESM diagnostic will be introduced. This work was set up in collaboration and financial support of Fusion for Energy.

Id 368

Abstract Final Nr. P1.017

A roadmap to the realisation of fusion energy: – mission for solutions on heat-exhaust systems

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The European fusion electricity roadmap sets out a strategy for a collaboration to achieve the goal of generating fusion electricity by 2050. It has been developed based on a goal-oriented approach with eight different missions including development of Heat-Exhaust systems which must be capable of withstanding the large heat and particle fluxes of a fusion power plant. The European Fusion Consortium (EUROfusion) strategy to set up an efficient Work Breakdown Structure and the collaborative efforts to address this challenge will be presented. The current baseline strategy to reduce the heat load on the divertor targets to secure acceptable divertor operation in the detached regime by radiating a sufficient amount of power from the plasma and by producing “detached” divertor conditions with sufficiently collisional SOL plasma will be pursued by studies based on existing, especially all metal plasma facing, divertor devices and supported by a strong model validation efforts. In such conditions a significant temperature gradient can be established and volume recombination of the plasma can take place, hence reducing the ion fluxes to the target. A specific Work Package (WP) will address the challenges of erosion of the plasma facing components in order to maximise the availability of the device and to reduce the deleterious effects of hydrogen co-deposition and dust production. Furthermore this approach will be tested by ITER, thus providing an assessment of its adequacy for DEMO. Nevertheless, the significant risk remains that high-confinement regimes of operation are incompatible with the larger core radiation fraction required and cannot be extrapolated to the DEMO fusion reactor, significantly delaying the realisation of fusion energy. The investment into risk mitigation strategy with Work Packages defining a viable alternative solution for heat exhaust on DEMO is required. This includes a technological study of feasibility and performance of water-cooled divertor targets concepts, which extend the ITER design and technology to DEMO relevant condition (e.g., higher coolant temperatures and pressures and higher n-dose). Design, assessment of the adequacy for DEMO and proof-of-principle tests of innovative geometries such as super-X and snowflake configurations as well as the use liquid metals are also addressed. Finally, as part of the coherent mission approach, the definition of the exact scope and technical specifications of a Divertor Tokamak Test (DTT) facility (either a new facility or the upgrade of existing facilities) will have to be completed and, after a thorough review, a decision should be taken in 2016 for its construction.

Id 535

Abstract Final Nr. P1.018

A compact high power density tokamak power supply

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This abstract uses Supercapacitors and IGBTs to supply power to multiple coils on a tokamak from a SINGLE capacitor bank.. This design is a novel compact high density power supply. Super capacitors combined with IGBTs allow high current switching to provide power with waveform current control at different currents to multiple coils using a single capacitor bank. This allows a more compact and high power density power supply system for tokamaks. Additionally this power supply has applications to industry to assist power supply balancing with active loads. On the ST25 tokamak the Toroidal Field and Poloidal Field coils have their own IGBTs which switch current from the main bank. A TF of 0.1T at a radius of 0.25m can be maintained for >6 second pulses. Using a microcontroller the bank voltage is monitored and PWM control of the IGBTs at approximately 1kHz is used to adjust the duty cycle in real-time to provide the desired current output. A power supply with a footprint of ~2 square metres consists of two Maxwell 125V transport modules in series providing 250V with an internal resistance of 36mO and capacity of 31.5F and capable of providing a current up to 2.4kA. This configuration can be paralleled up to provide more current. This paper describes the circuit and shows data where two coils have been simultaneously driven with different current waveforms from the same bank.

Id 1023

Abstract Final Nr. P1.019

Upgrade of ICRF heating system on EAST

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Ion cyclotron resonance frequency (ICRF) heating is one of the primary auxiliary heating techniques for Experimental Advanced Superconducting Tokamak (EAST). Recently, the ICRF system was upgraded to achieve long-pulse steady-state operation with ~12 MW power of 1000s in a frequency range of 25 to 70MHz. After construction of new four 1.5MW transmitters, The ICRF system, which consists of 8 transmitters, will give out ~ 12MW total power for the 2014 EAST experiment campaign. The power supply systems for drive power amplifier (DPA) and final power amplifier (FPA) of transmitter were upgraded by using reliable PSM high voltage sources, whose response time is less than 5 μs. Three new fast wave heating antennas at B port, F port and I port on EAST have been fablicated. Several projects have been developed for high-power operation such as cooling system for antennas and transmission network, vacuum feedthrough and feed-back control and protection system etc. The former 6.0MW (1000s) ICRF system has been put into operation and some encouraging ICRF experiment results have been obtained. By upgrading the transmitters, antennas, cooling system etc., the high-power, long-pulse, steady-state operarion capability of ICRF system on EAST has been promoted.

Id 334

Abstract Final Nr. P1.020

Key Enabling Design Features of the ITER HNB Duct Liner

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The Duct Liner (DL) for the ITER Heating Neutral Beam (HNB) is a key component in the beam transport system. Duct Liners installed into equatorial ports 4 and 5 of the Vacuum Vessel (VV) will protect the port extension from power deposition due to re-ionisation and direct interception of the HNB. Furthermore, the DL contributes towards the shielding of the VV and superconducting coils from plasma photons and neutrons. The DL incorporates a 316L(N)-IG, deep-drilled and water cooled Neutron Shield (NS) whose internal walls are lined with actively cooled CuCrZr Duct Liner Modules (DLMs). These Remote Handling Class 2 and 3 panels provide protection from neutral beam power. This paper provides an overview of the preliminary design for the component and focusses on critical features that ensure compatibility with: high heat flux requirements, remote maintenance procedures, and transient magnetic fields arising from major plasma disruptions. The power deposited on a single DLM can reach 300 kW with a peak power density of 2.4 MW/m2. Feeding coolant to the DLMs is accomplished via welded connections to the internal coolant network of the NS. These are placed coaxially to allow for thermal expansion of the DLMs without the use of deformable connections. Critically, the remote maintenance of individual DLMs necessitates access to water connections and bolts from the beam facing surface, thus subjecting them to high heat flux loads. This design challenge will become more prevalent as fusion devices become more powerful and remote handling becomes a necessity. The novel solutions implemented to overcome this are detailed and their performance scrutinized. The designs presented include tungsten caps that protect bolted and remotely welded connections by radiating heat away, and explosion bonded CuCrZr/Stainless Steel panels designed to reduce by a factor of 10 the eddy current torques caused by plasma disruptions.

Id 137

Abstract Final Nr. P1.021

High Heat Flux Engineering for the Upgraded Neutral Beam Injection Systems of MAST Upgrade

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MAST is undergoing a major upgrade which will see the installation of a highly flexible platform for divertor physics and will enable studies of effectively steady-state plasma performance. The upgraded MAST will be capable of 5 second plasma discharges driven initially by up to 5 MW of neutral beam injection (NBI). A major part of the present shutdown is to upgrade the NBI systems to increase their operational envelope to meet the expected performance of MAST-U. Based on operational experience and enhanced numerical ion-tracing analyses, the existing bending magnet and residual ion dump (RID) which were fit for MAST operations must be substantially upgraded to meet higher injected pulse energy and reliability requirements of MAST-U. This paper describes the modelling, design and procurement of a bespoke new bend magnet and RID, focusing on the challenges of high heat flux component design. The main design challenge is the limited space available in the beamline vacuum vessel. Studies have been performed using the in-house MAGNET code to track all residual ion species as well as re-ionised power. Additionally, the commercial OPERA code has been used to model the effects of incomplete space charge compensation and to validate the MAGNET results. The RID and magnet geometry were iteratively varied to deliver a compact final design with acceptable power density profiles peaking at 8-12 MW/m2. The resulting RID design uses a ‘V’ dump of CuCrZr HyperVapotron elements and a series of CuCrZr deep-drilled swirl tube plates. All new high heat flux surfaces have been analysed using thermo-structural analyses and are shown to pass the ITER SDC-IC design rules, including fatigue and the elasto-plastic route for ratcheting. The RID components are currently under manufacture and will be assembled ready for installation in late 2014.

Id 509

Abstract Final Nr. P1.022

Preparation for the next JET tritum campaign: performance of the EP2 PINIs with grid gas delivery

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The JET Neutral Beam Injection system consists of two injection boxes each having eight Positive Ion Neutral Injectors (PINIs). These EP2 PINIs operate with extracted deuterium beams up to 125kV, 65A resulting in a maximum injected neutral beam power of 2.2MW per PINI or ~35MW in total. The next JET tritium campaign will include operational phases where both injection boxes operate in tritium. Based on present operation, the predicted performance in tritium operation is 2.2MW of injected neutral beam power for an extracted ion beam power of 118kV, 45A;35MW in total . In deuterium operation the source operating pressure and neutraliser target are established by supplying gas directly into the ion source and also to a point approximately half way along the neutraliser. This system cannot be used for tritium operation since the gas delivery system does not have secondary containment in the event of a tritium leak. Engineering a high voltage ceramic break with secondary containment for the source gas is very challenging. Instead a special gas delivery system with secondary containment is used where both the source and neutraliser tritium gas is delivered at the earth grid of the accelerator (“grid gas delivery”). At the same total gas flow, the source pressure is less for grid gas delivery. Increasing the gas flow may lead to high voltage breakdowns. Hence it is important to verify the performance of the injectors with grid gas delivery. Measurements of PINI performance in both modes of gas delivery have been made on the Neutral Beam Test Bed for hydrogen and deuterium operation and the results compared. These show that in grid gas operation the arc efficiency is more dependent on the gas flow rate than in normal gas operation. A simple discharge model and conductance arguments are used to explain the results. This work, part-funded by the European Communities under the contract of Association between EURATOM/CCFE, was carried out within the framework of the European Fusion Development Agreement. This work was also part-funded by the RCUK Energy Programme under grant EP/I501045.

Id 610

Abstract Final Nr. P1.023

JET Neutral Beam Duct Optical Interlock

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The JET Neutral Beam Injection (NBI) system is the most powerful neutral beam plasma heating system currently operating. Optical Interlocks wereinstalled on the beam lines in 2011 for the JET Enhancement Project 2 (EP2), when the heating power was increased from 23 MW to 34 MW. JET NBI has two beam lines. Each has eight positive ion injectors operating in deuterium at 80 kV - 125 kV (accelerator voltage) and up to 65 A(beam current). Heating power is delivered through two ducts where the central power density can be more than 100 MW/m2. In order to deliver this safely, the beam line pressure should be below 2×10-5mbar otherwise the power load on the duct from the re-ionised fraction of the beam is excessive. The new Optical Interlock monitors the duct pressure by measuring the Balmer-alpha beam emission (656 nm). This is proportional to the instantaneous beam flux and theduct pressure. Light is collected from a diagnostic window and focused into 1-mm diameter fibres. The Doppler shifted signal is selected using an angle-tuned interference filter. The light is measured by a photomultiplier module with a logarithmic amplifier. The interlock activation time of 2 ms is sufficient to protect the system from a fully re-ionised beam – a significant improvement on the previous interlock. The dynamic range is sufficient to see bremsstrahlung emission from JET plasmas and not saturate during plasma disruptions. For high neutron flux operations the optical fibres within the biological shield can be annealed to 350 °C. A self-test is possible by illuminating the diagnostic window with a test lamp and measuring the backscatter. We demonstrate an important technology for the protection of high power neutral heating beams and present the design and operational results.

Id 549

Abstract Final Nr. P1.024

The mechanical structure of the WEST Ion Cyclotron Resonant Heating Launchers

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An important issue for the WEST (Tungsten (W) Environment in Steady-state Tokamak)project, which aims at modifying Tore Supra to an X-point divertor machine, is the operation of three Ion Cyclotron Resonant Heating (ICRH) launchers at a level of 9MW during 30s or 3MW during 1000s. The WEST ICRH system has to deal with two challenging issues that no other ICRH system before ITER has facedsimultaneously so far, i.e. ELMs resilience and Continuous Wave (CW) RF operation. The technical solution chosen to meet the requests imposed by the WEST scenariosis to build three new launchers based on the RF structure successfully tested in short pulses in 2007 on Tore Supra (TS) prototype launcher. The launcher RF matching circuit is modified and the generators are upgraded to cope with large reflected power and variations of coupling resistance. For the CW operation (3MW of injected power during 1000s), the design of the TS prototype has been upgraded, with a cooling of the launcher that requires to be integrated from the very beginning in the mechanical design. This paper gives an overview of the mechanical structure of the CW ELMs resilient WEST ICRH launchers. The technical solutions chosen to drive the mechanical design are presented, in regard of the past experience on the 2007 TS prototype, together with the significant work carried out on the mechanical design to improve the launcher structure. The thermal and electro-mechanical analyses conducted and their impact on the launcherdesign are also presented. The analyses and construction code recommendations on maximum allowable stresses and temperatures in the structure have driven the mechanical design. Indeed, the steady state operation, added to the high levels of constraints imposed by the heat fluxes (up to 2MW.m-2on some parts), the cooling water loop limits and the disruption forces strongly restrain the technical options. These three new CW ELMs resilient ICRH launchers are foreseen to be installed on WEST in 2016, and operational for the first plasmas.

Id 458

Abstract Final Nr. P1.025

R&D Activities on RF Contacts for the ITER Ion Cyclotron Resonance Heating Launcher

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Embedded RF contacts are integrated within the ITER ICRH launcher to allow assembling, shimming and lower the thermo-mechanical stress. They have to withstand a peak RF current of 2.25 kA at 60 MHz in steady-state conditions, under machine vacuum and temperature environment (up to 250°C) and during the whole life of the launcher without degradation. The RF contacts are critical components for the launcher performance and intensive R&D is therefore required, since no RF contact has so far been qualified at these specifications. In order to test and validate the anticipated RF contacts in operational conditions, CEA has prepared a test platform consisting of a steady-state vacuum pumped RF resonator. This resonator has been installed within the TITAN facility, a test stand facility compatible with the test of one ITER ICRH launcher module (1/4 of the launcher) and equipped with active water cooling including a hot pressurized water loop. In collaboration with ITER Organization and the CYCLE consortium (CYclotron CLuster for Europe), an R&D program has been conducted to develop a RF contact that meets the ITER ICRH launcher specifications. R&D on Cu/Ti brazing processes has been performed, as well as diffusion bonding tests between different Titanium grades. Commercial RF contacts have also been tested, but discarded following RF tests and brazing assembly failures. A design proposed by CYCLE consortium, using brazed lamellas supported by a spring to improve thermal exchange efficiency while guarantying high contact force, was tested successfully in the T-resonator up to 1.7 kA during 1200 s, but failed for larger current values due to a degradation of the contacts. Details concerning the manufacturing of the brazed contacts on its titanium holder, the RF tests results performed on the resonator and the non-destructive tests analysis of the contact are given.

Id 313

Abstract Final Nr. P1.026

Radio-Frequency electrical design of the WEST Long Pulse and Load-Resilient ICRH launchers

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The WEST project (Tungsten (W) Environment in Steady-state Tokamak) aims at modifying Tore-Supra to an X-point divertor tokamak, equipped with actively cooled tungsten Plasma Facing Units (PFU), with the goal to test divertor components technologies for ITER. In order to generate ITER-relevant high heat fluxes on its PFU, WEST will be equipped with three new Ion Cyclotron Resonance Heating (ICRH) launchers, designed to operate at 3MW/launcher for 30s and 1MW/launcher for 1000s on ELMy H-mode plasmas. These new ICRH launchers, optimized for Hydrogen minority heating scheme (f ~ 55 MHz) in dipole phasing configuration, will be up to date the first ICRH launchers to offer the unique combination of continuous wave operation, ELMs tolerance capabilities for coupling on H-mode edge, as well as reduced RF sheath potentials. In order to withstand the load variations caused by ELMs and to ensure a maximum Voltage Standing Wave Ratio of 2 at the generators, the design of the WEST ICRH launchers is based on the load-resilient concept consisting of a toroidal array of Resonant Double Loops with low junction impedance conjugate-T bridges (3Ω) and internal vacuum capacitors. This concept has already proven its load-resilience capabilities on both Tore-Supra and JET. The RF design optimization process is carried out using electromagnetic solvers (HFSS, COMSOL, TOPICA), combined with electric circuit calculators. The nominal frequency is optimized to 55MHz by lowering the straps reactance. This is achieved by reducing the mechanical and electrical lengths of the straps, compared to the ITER-relevant prototype ICRH launcher tested on Tore-Supra. In addition, the launchers front-face (strap geometry and dimensions, Faraday Screen) is optimized to improve plasma-antenna coupling, reduce RF sheath potentials as well as to increase the power handling capabilities. The launchers rear part (bridge and the impedance transformer) is optimized to improve the matching, load-resilience and power handling.

Id 60

Abstract Final Nr. P1.027

Evolution of the Tore Supra Lower Hybrid Current Drive System for WEST

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The WEST-project (W-tungsten Environment in Steady-state Tokamak) involves equipping Tore Supra with a full tungsten divertor capable of withstanding heat flux load in the 10-20 MW/m2 range in steady-state conditions, with the aim to test W-divertor technologies. Long pulse operation, including full non-inductive scenarios, will be sustained by Lower Hybrid Current Drive (LHCD). The LHCD generator has recently been upgraded to deliver 9MW/1000s, and is now equipped with sixteen TH2103C klystrons, each with a power capability of 620 KW/1000s on VSWR=1.4. Half of the generator (eight klystrons) has been successfully commissioned on Tore Supra limiter plasmas, with the Full Active Multijunction (FAM) launcher, reaching RF power of 3.8 MW coupled to the plasma. The other part of the generator, so far commissioned on matched load, will be tested with the Passive Active Multijunction (PAM) launcher at the start of WEST operation in 2016. The LHCD launchers need significant modifications to adapt to the WEST plasma geometry. In particular, the WEST transformation involves reducing the plasma volume, thus moving the launchers ~10cm closer to the tokamak centre. The toroidal curvature of the launchers no longer fits the plasma curvature, due to the strong magnetic field ripple effect. This leads to a degradation of the LH wave coupling, as studied in dedicated Tore Supra experiments simulating WEST configuration. It was observed that, due to their different coupling characteristics, the coupling on the FAM was strongly degraded in WEST-like plasmas, while the coupling on the PAM remained rather unaffected. It was therefore found mandatory to reshape the toroidal curvature radius of the FAM launcher mouth from 1700mm to 2300m. The machining of the front face of the FAM launcher is an irreversible process and the risk is minimized by shaping a full size mock-up, as described in this paper.

Id 357

Abstract Final Nr. P1.028

Validation on Test Bed of the Tore Supra Electron Cyclotron Launcher Upgrade

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This paper presents the results of the validation on the test Bed of the Tore Supra Electron Cyclotron Launcher Upgrade. This upgrade is consisted in mechanical and real time control system modifications in order to increase the accuracy of the control of the wave injection angles, the speed and the reliability. The new system of this launcher was tested on the test bed in vacuum and high temperature conditions (in a simulated tokamak environment) to confirm the required precison, the reliability and the repeatability. The test validation process described in this paper could also be applied to ITER EC launchers.

Id 388

Abstract Final Nr. P1.029

Commissioning of the 28 GHz ECRH power transmission line for the TJ-II stellarator

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In the accompanying text, the commissioning of the 28GHz power transmission line of the TJ-II stellarator designed for the excitation of Bernstein waves will be presented. In the first place, the original design, deployment and layout of the corrugated waveguide (including the necessary bends), gyrotron, Matching Optics Unit (MOU) and vacuum windows will be described. These topics will lead us to the coupling problems originated by a MOU in which spillover effects were not rigorously enough taken in to account. This problem made the beam not to correctly fit to the waveguide, therefore exciting higher order modes that in the long run ended provoking the physical breakdown of the vacuum window. In order to clarify this part, figures showing infrared (IR) measurements on the window will be presented, which will help identifying higher order modes and the coupling problem. In order to get rid of these hindrances, another MOU was projected, designed and installed. An additional set of thermographical measurements helped to visualize that the beam was correctly fitting the waveguide, therefore not exciting higher modes due to beam size mismatch. Nevertheless measurements showed higher order modes which led to the conclusion that alignment mismatch effects were present and led to a more precise optic (low power)-high power alignment. In the meanwhile, polarization measurements were carried out in order to verify that the correct polarization was being launched to the vacuum vessel. After realizing the corrugated waveguide bends were introducing a very important change in the desired polarization, theoretical studies were performed in order to estimate it. That also led to the design and measurement of the grooved mirrors needed to achieve the final polarization. These works concluded with the actual setup and different measurements of the power line which will also be presented.

Id 647

Abstract Final Nr. P1.030

Design of the ITER NBI Passive Magnetic Shield

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The neutral beam system for ITER consists of two heating and current drive neutral ion beam injectors (HNB) and a diagnostic neutral beam (DNB) injector. The proposed physical plant layout allows a possible third HNB injector to be installed later. The Passive Magnetic Shield (PMS) works in conjunction with the active compensation/correction coils to limit the magnetic field inside the Beam Line Vessel (BLV), Beam Source Vessel (BSV), High Voltage Bushing (HVB) and Transmission Line (TL) elbow to acceptable levels that do not interfere with the operation of the HNB components. This paper describes the current status of the design of the PMS. The proposed vessel and TL PMS design was approved with minor comments during the Preliminary design review held in ITER organization in April 2013. The bushing PMS design has evolved since then due to changes regarding the position of the interface with the HVB. The paper focuses on the description of the structural layout and assembly strategy for the PMS. The PMS is an assembly of low carbon steel plates surrounding the vessel, high voltage bushing and transmission line elbow. For the vessel, the PMS is box like bolted assembly of panels made of two 75mm thick S235 low carbon steel PMS plates separated by a 100mm thick stainless steel internal frame. For the TL and bushing, the PMS is made of welded 150mm S235 plates, some of them curved into cylindrical shapes. The assembly strategy for the vessel PMS needs to minimise as much as possible the gaps between the PMS panels, as these have a negative effect on the magnetic shielding function of the vessel PMS. The calculations of field inside the PMS have been carried out assuming 1mm gaps between vessel PMS panels. The assembly strategy described is based on preassembly at the manufacturer's site and use of positioning pins to reproduce relative positions between panels during assembly in ITER.

Id 870

Abstract Final Nr. P1.031

Magnetic analysis of the Magnetic Field Reduction System of the ITER neutral beam injectors.

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The neutral beam system for ITER consists of two heating and current drive neutral ion beam injectors (HNB) and a diagnostic neutral beam (DNB) injector. The proposed physical plant layout allows a possible third HNB injector to be installed later. For the correct operation of the beam, the ion source and the ion path until it is neutralized must operate under a very low magnetic field environment. To prevent the stray ITER field from penetrating inside those mentioned critical areas, a Magnetic Field Reduction System (MFRS) will envelop the beam vessels and the high voltage transmission lines to ion source. This system comprises the Passive Magnetic Shield (PMS), a box like assembly thick low carbon steel plates, and the Active Correction and Compensation Coils (ACCC), a set of coils carrying a current which depends on the Tokamak stray field. This paper describes the magnetic model and analysis results presented at the PMS and ACCC preliminary design review held in ITER organization in April 2013. The paper focuses on the magnetic model description and on the description of the analysis results. The magnetic model includes a sector of the tokamak coils and air enveloping the PMS and ACCC. The simplifications from the real model and the modelled gaps between PMS panels are described. Gap width is simulated in the model by changing the magnetic permeability of the gap volume. The coil current optimization algorithm is based on an optimization function provided by ANSYS. The iterative process for obtaining optimized currents in the coils is presented. The chosen set of coils currents chosen among the many possible solutions, the magnetic field results in the interest regions and the fulfilment of the magnetic field requirements are described.

Id 871

Abstract Final Nr. P1.032

Structural analysis of the Passive Magnetic Shield for the ITER Heating Neutral Beam Injector system

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The neutral beam system for ITER consists of two heating and current drive injectors (HNB) and a diagnostic neutral beam (DNB) injector. The proposed physical plant layout allows a possible third HNB injector to be installed later. The PMS main function is to shield the injector from the external magnetic field coming from the tokamak and to shield the NB cell from the radiation coming from all activated components. The magnetic shielding is performed in association with the active compensation/correction coils (ACCC) to limit the magnetic field inside the Beam Line Vessel (BLV), Beam Source Vessel (BSV), High Voltage Bushing (HVB) and Transmission Line (TL) elbow to acceptable levels that do not interfere with the operation of the HNB components. The aim of the analysis described in this paper is to verify that the preliminary design of the Passive Magnetic Shield fulfils RCC-MR code requirements in case of seismic event in normal operation and during NB pulse. The analysis has been performed by a sequence of a modal analysis, followed by a static structural analysis in which the applied load is the gravity plus the pressure load on the vessel (transmitted to the PMS through the vessel supports), the electromagnetic load and the acceleration calculated for every seismic event. Stress predictions have only been linearized when RCC-MR limits have been exceeded when considering the total Von Mises equivalent stress. In all other cases the total equivalent stress alone has been used. This is a conservative approach which was deemed acceptable at this stage of design.

Id 960

Abstract Final Nr. P1.033

Design of the ITER NBI Active Correction and Compensation Coils

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The neutral beam system for ITER consists of two heating and current drive injectors (HNB) and a diagnostic neutral beam (DNB) injector. The proposed physical plant layout allows a possible third HNB injector to be installed later. For the correct operation of the beam the ion source and the ion path until it is neutralized must operate under a very low magnetic field environment. To avoid the stray ITER field to penetrate inside those mentioned critical areas, a Magnetic Field Reduction System (MFRS) will envelop the beam vessels and the high voltage transmission lines to ion source. This whole system comprises the Passive Magnetic Shield (PMS), a set of thick steel plates, and the Active Correction and Compensation Coils (ACCC), a set of coils carrying a current which depend on the Tokamak stray field. The ACCC are located up and down the beam path, three in the upper part and three in the lower one. This paper describes the status of the coil design presented at the Preliminary Design Review in ITER organization in April 2013. The coils are manufactured from deoxidized copper profiles insulated with pure fibre glass tapes and vacuum impregnated with epoxy resin. All the coils are components suitable to be removed from their final position to be replaced in the case of a fault in any part of them. Two upper coils, basically covering the space over the beam neutralizer and the ion beam dump, need to be fully disassembled by remote handling procedures in the case of any maintenance task inside the beam line vessel. The detail design, turn distribution, electrical and hydraulic terminals, support structures, structural and thermal calculations and basic operations in the case of disassembly are described in the paper.

Id 955

Abstract Final Nr. P1.034

Loads due to Stray Microwave Radiation in ITER

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High-power microwaves will be extensively used in ITER for a variety of purposes such as assisting plasma breakdown, plasma heating, current drive, tearing mode suppression and as a probing beam for the Collective Thomson Scattering diagnostic. In a number of these schemes absorption of the microwaves by the plasma will not be full and in some cases there could be no absorption at all. This may result in a directed beam with a high microwave power flux or - depending on location and plasma conditions - an approximately isotropic microwave power field. In both situations this may lead to unwanted and potentially harmful absorption by in-vessel components or to damage to diagnostic detectors. This paper assesses microwave stray radiation loads inside the ITER vessel by following an initial number of passes of the directed beam but also by viewing the ITER vessel as a microwave resonator. The effects of the loads are addressed by reviewing microwave heating of different types of components analytically and in some cases back-up by dedicated measurements. Finally a bolometer that can discriminate microwave heating from other heat sources of the plasma is discussed.

Id 603

Abstract Final Nr. P1.035

From the conceptual design to the first mock-up of the new WEST plasma control system

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The Tore Supra WEST project aims at the technology testing of one key component of the ITER project, namely its actively cooled tungsten divertor, for mitigation of the risks related to ITER operation. This new configuration leads to control issues and event handling close to that of ITER from a plasma scenario point of view (X-point configuration, H mode, long duration pulse) and from a machine protection point of view (metallic environment). Based on previous conceptual studies and to fill the WEST requirements, a dedicated sub-project aims to implement a new plasma control system (PCS) and a new pulse schedule editor (PSE). The main idea is to use a segment approach to describe the pulse scheduling with a full integration of event handling both on the PCS and on the PSE. After detailed specification work, it has been found that the real-time framework called DCS (Discharge Control System) which is currently used on ASDEX upgrade will be the best option to build the WEST PCS. For the PSE, the Xedit tool, developed for the future W7X facility, has been chosen. This contribution will begin by a short reminder on the concepts proposed for the control of the plasma and the handling of events during plasma discharge. Then it will focus on the new centralized architecture of the new Tore Supra PCS and its first mock-up based on a Multi-Inputs/Multi-Outputs plasma current and loop voltage control. This later will illustrate the required modifications of DCS and Xedit to fit with the Tore Supra Control infrastructure: communication between DCS and the other control/acquisition units will be detailed and along with Xedit adaptations to the WEST PCS segment concepts. An explanation of how Xedit will fill the DCS configuration files will be provided. After this validation phase, the future work will be to integrate more and more functionalities to this mock-up to reach at the end all the requirements for the WEST operation.

Id 193

Abstract Final Nr. P1.036

CREATE-NL+: a Robust Control-Oriented Free Boundary Dynamic Plasma Equilibrium Solver

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CREATE-NL+ is a FEM (Finite Elements Method) solver of the free boundary dynamic plasma equilibrium problem, i.e. the MHD time evolution of 2D axisymmetric plasmas in toroidal nuclear fusion devices, including eddy currents in the passive structures, and feedback control laws for current, position and shape control. This is an improved version of the CREATE-NL code developed in 2002 which was validated on JET and used for the design of the XSC (eXtreme Shape Controller), used for simulation studies on FTU, MAST, ASDEX-U, ITER, FT3 and DEMO. A significant improvement was the use of a robust numerical scheme for the calculation of the Jacobian matrix within the Newton based scheme for the solution of the FEM nonlinear algebraic equations. The improved capability of interfacing with other codes, and a general decrease of the computational burden for the simulation of long pulses with small time steps makes this code a flexible tool for the design and testing of magnetic control in a tokamak. The code is integrated within a suite of tools allowing the optimiziation of nominal preprogrammed currents to drive the plasma through a desired scenario and the design of closed loop control laws for shape control and vertical stabilization. Some of the latest experiences in simulating ITER and DEMO scenarios are presented to illustrate the code capabilities.

Id 660

Abstract Final Nr. P1.037

Improvement of 2D plasma identification by preliminary treatment of 3D measurement

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Axially-symmetric plasma equilibria are frequently altered by several 3D elements including active coils, vessel and first wall geometry [1]. Note that also flux and field probes, fastened to vessel or mechanical supporting structures, are influenced by any lack of symmetry. Plasma position and shape identification are mainly based on magnetic measurements. As a consequence, it is highly advisable to perform a preliminary signal treatment aimed to identify and reject, as much as possible, spurious effects due to possible lack of symmetry and, in addition, trying to recover equivalent measurements in axially-symmetric configuration. The paper deals with this problem by introducing, in order to describe the actual probes position, a description base for possible 3D geometries, characterized by small deformations with respect to nominal (axially-symmetric) configuration, examples being shifts or rotations along orthogonal axes. The projection of the measured data in the deformation base provides hints about possible deformations of the geometry and then an equivalent set of measurements is recovered by subtracting the effect of deformation. In addition, the procedure takes advantage from a comparative analysis between measurements with or without plasma. The paper will present a possible problem formulation and its implementation in a parallel computing environment, suited for real time identification [2]. A parametric analysis will be used to show the actual relevance of the issue in practical applications. This work was supported in part by Italian MIUR under PRIN grant 2010SPS9B3. [1] J. Knaster et al., Fus. Eng. Des. 86 (2011) [2] A. G. Chiariello. A. Formisano, R. Martone, Fus. Eng. Des. 88 (2012)

Id 984

Abstract Final Nr. P1.038

Effective Magnetic Field Computation in Tokamaks in Presence of Magnetic Materials

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Ferromagnetic materials play an important role in the magnetic fields distribution inside Tokamaks. In some specific applications, there is the need for a prompt and sufficiently accurate computation. A typical example is the impact of ferromagnetic parts on the magnetic measurements used as data source in the plasma shape and position reconstruction. In such case, starting from probe signals and knowledge of external currents, the information about plasma must be recovered by filtering out the contribution of magnetic materials and eddy currents in passive structures. For real time application, it is mandatory to achieve estimates of relevant parameters in times suited for plasma control. This paper proposes a set of simplified models for the treatment of ferromagnetic parts, based on equivalent sources characterized by reduced complexity. As a matter of fact, the magnetic material can be represented using a set of dipoles, and analytical formulas used for evaluating the contribution to the magnetic field. Magnetic moments associated to each dipole are non-linearly dependent on the actual magnetic field; an iterative procedure is then proposed to evaluate the actual momentum of each dipole. In order to speed up the process, advantage can be taken from a preliminary evaluation, even if rough, of the solution to be used as a initial guess. In this way, the contribution of the magnetic material to the vector potential and flux density is separately computed and suitably added to the one generated by the active sources. The model efficiency and accuracy are validated against accurate numerical codes using fine meshes. The adoption of high performance computing architectures is also considered to improve promptness. This work was supported in part by Italian MIUR under PRIN grant 2010SPS9B3.

Id 984

Abstract Final Nr. P1.039

A data-based model for Thermal SHAx prediction in RFX mod

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In the reversed field experiment RFX-mod, the operation at high plasma currents (Ip≥ 1.2 MA) has evidenced the formation of large electron thermal structures, delimited by strong Internal Transport Barriers (ITBs), in the plasma core during the quasi-single helicity (QSH) regimes. These structures appear when a helical equilibrium with a single helical axis (SHAx) has been established in the plasma. Despite the strong theoretical and experimental effort some aspects of the ITBs physics are still unclear. One of the open questions is why the ITBs crash in spite of a longer magnetic QSH state. As the presence and the class of ITBs is now recognized with a post-shot investigation of the electron temperature profiles, the aim of the work is to realize a predictive system, based on a data-based approach, able to automatically recognize the nature of ITBs and the possible precursor of their crashing phase. The predictive strategy consist in three phases: in the first one a database is built to train the predictor relying on signals belonging to double filter multi-channel diagnostics; the second realizes a clustering of measurements belonging to the same class of ITB by means of linear programming, while the third performs a projection of the clusters into a low-dimensional space, by means of a probabilistic neighborhood function, in order to recognize safe and crashes areas. Those areas are then used to test new sets of signals while the low-dimensional space allows better results visualization. The procedure has been tested on a subset of shots, including hydrogen and deuterium campaign from 2011 to 2014, and in this paper the results are presented.

Id 415

Abstract Final Nr. P1.040

Integration of Simulink, MARTe and MDSplus for rapid development of real-time applications

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Simulink is a graphical data flow programming tool for modeling and simulating dynamic systems and is widely used in the fusion community for the development of algorithms for plasma control. A component of Simulink, called Simulink Coder, generates C code from Simulink diagrams.Code generated in this way has already been integrated in several plasma control systems. MARTe is a framework for the implementation of real-time systems, currently in use in several fusion experiments. MARTe is based on an abstraction of the underlying operating system and provides an execution model where a number of real-time threads execute each a set of Generic Application Modules (GAMs). GAM instances will refer to input/output components as well as control algorithms. MDSplus is a framework widely used in the fusion community for the management of data. MDSplus allows storing the configuration of an experiment in a database called Experiment Model that is then cloned and filled with acquired data to produce a new database, called Pulse File, containing the complete picture of the configuration and the outcome of a given plasma discharge experiment. The three systems provide a solution to different facets of the same process, that is, real-time plasma control development. Simulink diagrams will describe the algorithms used in control, which will be implemented as MARTe GAMs and which will use parameters read from and produce results written to MDSplus pulse files. The paper will present an integration of the three systems suitable to speed up the development of real-time control applications. In particular, it will be shown how from a Simulink diagram describing a given algorithm to be used in a control system, it is possible to generate in an automated way the corresponding MARTe and MDSplus components that can be assembled to implement the target system.

Id 416

Abstract Final Nr. P1.041

A boundary integral method for eddy-current problems in fusion devices

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An accurate control of the magnetic boundary of a thermonuclear plasma is a crucial issue. Close-fitting passive conducting structures are an efficient way to prevent the growth of MHD instabilities, but are not suitable for a steady state fusion reactor, because of the finite diffusion time of any material. Nevertheless, given the relatively slow growth rate of RWMs, feedback control by means of active coils is possible. In practice, given the characteristics of both the target instability and the active control scheme, successful experiments and modelling take advantage of a careful knowledge of the effects of the 3D conducting structures surrounding the plasma [1]. In fact, in presence of closed loop control actions, 3D eddy currents and plasma perturbations will evolve in a coupled way. In this work, we present an effective technique to solve eddy current problems in thin conductors of arbitrary topology by a Boundary Element Method (BEM) based on a stream function. The aim of this paper is to introduce a novel technique to render the stream function single valued when the thin conductor is not topologically trivial (e.g. toroidal surfaces with holes). In particular, a novel combinatorial algorithm to compute the appropriate cohomology generators in linear time worst case complexity is introduced, providing an effective and rigorous solution for the required topological pre-processing. The proposed technique is applied to study, in frequency domain, the dynamic response of the MHD control system of RFX-mod, a medium size (R=2m, a=0.459m) device, equipped with a complicated (geometrically and topologically) stabilizing shell and a state of the art feedback system. [1] P. Bettini, L. Marrelli, and R. Specogna. Calculation of 3D magnetic fields produced by MHD active control systems in fusion devices. IEEE Trans. on Magnetics, 50 (2014), 7000904

Id 352

Abstract Final Nr. P1.042

Feasibility study of a local active correction system of magnetic field errors in RFX-mod

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The active control system of MHD modes is a key feature of RFX-mod device and it allowed performing a variety of control experiments operating the machine both as a high current RFP and a low current, low q Tokamak. Expertise in the active control of plasma modes along with the highest accuracy in the equilibrium magnetic field configuration are stringent requirements in ITER operation scenarios. In RFX-mod, studies are under way to review the existing load assembly with the aim of further increasing the quality of the magnetic configuration. In spite of its overlapping edges, the poloidal cut of the copper shell still remains one of the major sources of field error. On the basis of the experience acquired in the active control, a feasibility study of a local correction system made up of a set of dedicated coils has begun. It includes the development of a cylindrical FE model of copper shell, stainless steel support structure, local coils, existing MHD saddle probes and new local ones. The frequency characterization of the system response provides basic information for checking the system effectiveness in producing the desired magnetic field at both the probes and the plasma boundary. Moreover, it provides data for the identification of a dynamic model whose inputs are the local coil currents and whose outputs are the saddle probe magnetic field measurements. In particular, a map of the magnetic field components at the saddle probes radius is necessary to evaluate the spatial harmonic content of the local correction system and subsequently to analyse its interaction with the 192 saddle coils covering the whole support structure. Conversely, the transfer functions between neighbouring saddle coils and local system probes are studied. Different decoupling strategies of the two control systems are considered both in terms of harmonic components and raw signals.

Id 562

Abstract Final Nr. P1.043

Strategies for real-time actuator decoupling in closed-loop MHD control operations

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In many devices aiming at magnetic confinement of fusion relevant plasmas, feedback control of MHD instabilities by means of active coils is nowadays mandatory to ensure the robustness of high performance operational scenarios. Actuators involved in the control loop are often coupled in the sensor measurements and an optimal strategy for decoupling can be limited by the need of reducing as much as possible the cycle time of the control loop itself. RFX-mod is a medium size (R=2m, a=0.459m) device able of confining plasmas in both Tokamak and Reversed Field Pinch magnetic configurations. It is equipped with an advanced feedback system for field error and MHD control [1]. Actuators in this system are 192 active saddle coils entirely covering the plasma outer surface, while more than 600 magnetic sensors are included in the control loop, providing the operator with a challenging coupling situation. It is also important to stress the fact that the problem is intrinsically 3D, involving different non-axisymmetric contributions. All these characteristics can in principle be included in a (complex, frequency dependent) mutual coupling matrix that, when properly inverted and implemented in the control loop, should provide the necessary corrections to the actuator action. We will start by documenting the baseline situation in RFX-mod, where the Identity matrix is chosen to represent the simplest case of mutual coupling matrix. After that, implementation and use of decoupling matrices for the two limiting cases of zero and infinite frequency will be presented. The issue of properly modelling the problem will be also tackled by comparing results from white and black-box approaches

Id 577

Abstract Final Nr. P1.044

Development of an ITER Prototype Disruption Mitigation Valve

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Disruptions in tokamaks seem to be unavoidable. Consequences of disruptions are (i) high heat loads on plasma facing components, (ii) large forces on the vacuum vessel, and (iii) the generation of runaway electron beams. In ITER the thermal energy of the plasma needs to be evenly distributed on the first wall in order to prevent melting, forces from vertical displacement events have to be minimised, and the generation of runaway electrons should be suppressed. Massive gas injection using fast valves is a concept for disruption mitigation which is presently explored on many tokamaks. Fast disruption mitigation valves based on an electromagnetic eddy current drive have been developed in Jülich since the 90's and models of various sizes have been built and are in operation on the tokamaks TEXTOR, MAST, and JET. A disruption mitigation valve for ITER is necessarily larger with an estimated injected gas amount of ~20 kPa.m3 for runaway electron suppression and all used materials have to be resistant against much higher levels of neutron and gamma radiation than on existing tokamaks. During the last five years the concept for an ITER prototype disruption mitigation valve has been developed up to the stage that an fully functional valve could be build and tested. Although the principal design is similar to the valves developed so far, special emphasis had to be put on the development and functional testing of a couple of critical items: (i) the injection chamber seal, (ii) the piston seal, (iii) the eddy current drive, and (iv) a braking mechanism to avoid too fast closing of the valve which could damage the gas volume seal. The concept of the valve will be introduced and detailed solutions and testing results for the critical items will be presented.

Id 546

Abstract Final Nr. P1.045

The Remote Control System for EAST Tokamak

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The international collaboration becomes popular in the research field of Tokamak because the experiment facilities are larger and more complex than the former ones. The traditional on-site collaboration model, which has to spend much money and time on international travel, is not suitable for the more frequent international collaboration of Tokamak research. The Experimental Advanced Superconducting Tokamak (EAST), which was constructed by the Institute of Plasma Physics Chinese Academy of sciences (CASIPP), is also facing the same problem. The collaborations between CASIPP and laborites, institutes all over the world are working well and much closer than ever before. It is essential to find a low cost and high performance alternative to traditional international collaboration model. The main objective of Remote Control System (RCS) for EAST is to provide an integration platform that allow collaborators to access experiment data, monitor system status, control the EAST facility remotely and communicate with each other. This paper presents a description of the main design goals, the key technical issues and implementation considerations for EAST Remote Control System.

Id 286

Abstract Final Nr. P1.046

Design and realization of data Management System in technical diagnosis system of the EAST superconducting magnet system

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Technical diagnosis system (TDS) is one of the important subsystems of EAST (experimental advanced superconducting tokamak) device. In EAST nuclear fusion experiment, TDS monitors real-time thermal hydraulic and electric parameters of the superconducting (Abbr. SC) magnet system in order to monitor the status parameters and to maintein the operating stability and security especially during each plasma discharging. It plays an important role of design of a high efficient data management system in EAST TDS. TDS data management scheme basically depends on three key factors in nuclear fusion experiment. First, there is demand for both local and remote data monitoring and storage. Second , function requirements like real-time visualization, online analysis and longtime waveform presentation are required. The third factor is the properties of TDS data objects. From data management perspective, TDS data objects are mainly divided into three types composed of steady-state data and pulse data and random data. Therefore, two-layer data management system is designed with both local and remote databases to satisfy different usages and real-time data processing demand. In the meantime four-type databases are applied in TDS data management system. Microsoft access database is used for configuration and arithmetic operation of signal input channels. Microsoft SQL Server database is chosen for long-time steady-state data storage. My SQL database is adopted for real-time data web presentation. MDSplus database also is adopted for online data analysis and long-pulse data waveform presentation. In the paper it is discussed that how to design and realize TDS data management system. Key difficulties during the implementing process are put to highlights. Future development is also introduced. It proved in past experiments that TDS data management system cannot only deal with large volume data efficiently but also satisfy kinds of function requirements.

Id 82

Abstract Final Nr. P1.047

Calculation of neutron hardness factors and gamma doses for the estimation of the radiation damage of the BES system of the EAST tokamak

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In this work we present a powerful method to characterise the expected radiation loads of the cameras of the BES system on the EAST tokamak predicting their behaviour in radiation fields and operation lifetimes. A BES system was designed and installed to the KSTAR tokamak in cooperation between the Wigner RC and BME. During the campaigns of the KSTAR tokamak radiation induced performance degradation has been experienced of the cameras of the BES system. Neutron and photon transport calculations were carried out with MCNP to calculate the neutron fluence, spectrum and gamma dose in the camera postions. The MCNP model was made by MCAM which allows minimal difference beetween the CATIA model and the MCNP model. According to the Lindhard's theory, the neutron hardness factors and the 1 MeV equivalent neutron fluence in silicon were calculated from the calculated neutron spectra. Owning of the hardness factors and the radiation damage of the cameras on the KSTAR tokamak opens the oppurtunity to predict the BES system performance on the EAST tokamak. The expected operation lifetimes can be calculated of the cameras, moreover the hardness factors also gives the possibility to design new and efficient neutron shielding to extend the lifetime of the cameras.

Id 857

Abstract Final Nr. P1.048

Provenance metadata gathering and cataloguing of efit++ equilibrium code execution

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Journal publications, as the final product of research activity, are the result of an extensive and complex modelling and data analysis effort; it is of paramount importance, therefore, to capture the origins and derivation of the published data in order to achieve high levels of scientific reproducibility, transparency, internal and external data reuse and dissemination. The consequence of the modern research paradigm is that high performance computing and data management systems, together with metadata cataloguing, have become crucial elements within the nuclear fusion scientific data “lifecycle”. This paper describes an approach to the task of automatically gathering and cataloguing provenance metadata, currently under development and testing at Culham Center for Fusion Energy. The platform is being applied to a complex intershot plasma equilibrium code EFIT++ as a proof of principle test. The proposed approach avoids any code instrumentation or modification. It is based on the observation and monitoring of input preparation, workflow and code execution, system calls, log file data collection and interaction with a subversion control system. Pre-processing, post-processing, and data export and storage are monitored with the code runtime. Input data signals are captured using a data distribution platform called IDAM. The performance, overhead and scalability of the proposed approach will be discussed. The final objective of the catalogue is to create a complete description of the modelling activity, including user comments, and the relationship between data output, the main experimental database and the execution environment. A prototype web interface for catalogue interrogation, visualization and reuse of the data will be presented.

Id 753

Abstract Final Nr. P1.049

The JET Neutron Calibration 2013 and its Results

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The power output of fusion experiments and fusion reactor-like devices is measured in terms of the neutron yields which relate directly to the fusion yield. In this paper we describe the methods used to make the in-situ calibration of JET in April 2013 and its results. We met the target accuracy ofthis calibration which was 10%, as in the earlier JET calibration and as required for ITER, where a precise neutron yield measurement is important, e.g. for tritium accountancy. We calibrated the two main systems which carry the JET calibration, ie the external Fission Chamber detectors and the Activation System. This was the first direct calibration of the Activation system in JET. We used the existing JET remote-handling system to deploy the 252Cf neutron source and developed the compatible tooling and systems necessary to ensure safe and efficient deployment in these cases.The scientific programme has sought to better predict and understand the limitations of the calibration, to optimise the measurements and other provisions, to provide corrections for perturbing factors (e.g. presence of the remote-handling boom and other non-standard torus conditions) and to ensure personnel safety and safe working conditions. Much of this work has been based on an extensive programme of Monte-Carlo calculations. Examples are given. The analysis of the Activation data leads to a change in the JET calibration of ~ 16% and the Fission chamber data analyses agree with this finding. Cross-calibrationdata have since been obtained which allow the calibration experiment results to be applied to JET D,D plasma pulses. The result is that the JET pulse outputs have been historically underestimated after about 2003, and should be adjusted upwards by ~16%. Unfortunately, we do not have sufficient records of the earlier physics or engineering details to allow modelling to ascribe earlier adjustments with certainty.

Id 1035

Abstract Final Nr. P1.050

RAMI approach as guidance for the design of the WEST machine protection system using IR thermography measurements

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The WEST project (Tungsten (W) Environment in Steady State Tokamak), is targeted at minimizing risks in support to the ITER divertor strategy, bringing together tungsten actively cooled component technology and tokamak environment by installing a full W actively cooled divertor to operate over long plasma discharges. Part of the machine protection system will be based on Infra-Red (IR) thermography which consistsin monitoring and controlling in real time the power load on the Plasma Facing Components (PFCs) through the surface temperature measurements. A high inherent availability value of such a machine protection diagnostic is essential for WEST operation and investment. The initial IR system consisted of a set of 3 different diagnostics: six cameras located in upper ports viewing the full W divertor, five novel viewslocated behind the inner protection panels for the antennas monitoring and based on an innovative imaging fibers bundle technology and a tangential wide angle view located in a median port, for the upper divertor target and first wall monitoring. The initial design of the system has been analyzed using a Reliability, Availability, Maintainability and Inspectability (RAMI) approach. A functional analysis of the IR thermography system from highest level functions down to basic operational functions has been developed to be able to estimate the potential reliability and availability of such a protection system. Results showed that the five fibers bundle system is the most critical components due to a weak reliability and a poor accessibility which would lead to a downtime for repairing a failure of at least one month. Despite mitigation actions to reduce the frequency of potential failures and the time to repair them, the availability required by the project could not be reached for this bundle system. With the aim of achieving the availability target, an alternative design concept has been developed. The new design which uses in-vessel convex mirrors inserted in the inner first wall to transmit photons from RF antennas towards the upper ports, instead of fibers bundle used in the initial design, allows increasing both the reliability and the time to repair of the optical system. This paper presents the RAMI analysis of the IR thermography diagnostic whose results have allowed guiding the design of the system used to protect the RF antennas to a more reliable solution as required by the WEST project.

Id 259

Abstract Final Nr. P1.051

Enhanced Integrators for West Magnetic Diagnostics

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On Tokamaks, the magnetic diagnostic provides some of the most important contributions to the plasma control, plasma physics and machine protection. Accurate and real time measurements are then required to ensure good plasma localization and control, all along the pulse. In hard tokamak environment, most of the magnetic sensors are based on the principle of pick-up coils that are sensitive to the magnetic field variation (dB/dt), so that the magnetic field (B) measurement requires the use of an integration system. Analogue integrators are one of the ways to achieve this kind of measurements. Since long time Tore Supra magnetic diagnostic uses analog integrators and electronics team gathered a large experience with such kind of electronic modules, especially during very long pulses. Even though analog integrators are subject to intrinsic issues, like drift, they proved during years that they are accurate, reliable, not expensive and above all, real time working. The West project includes a divertor and then more magnetic sensors than last version of Tore Supra did. As West project needs several tens of additional integrators, this opportunity has been held to consider an enhanced design for the new integrator modules. Three prototypes have been designed, using different combinations of symetrical integrators, isolated power supply and digital drift compensation system, aiming to achieve lower drift (<50μV.s over 1000s), larger common mode rejection ratio (CMRR >100dB) and common mode voltage range. The paper presents the explored and tested ways we followed in order to create enhanced analogue integrators for the West project. The results of extensive and exhaustive tests of the three different prototypes are summarized. The achieved performances are analyzed regarding the West project specifications and furthermore, the last ITER requirements for magnetic diagnostics integrators are also considered.

Id 263

Abstract Final Nr. P1.052

Design of soft-X-ray tomographic system in WEST using GEM detectors

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In metallic Tokamaks, the interplay between particle transport and MagnetoHydroDynamic (MHD) activity might lead to impurities accumulation and finally to disruption. Studying such phenomena is thus essential if stationary discharges are to be achieved. Measuring the Soft X-Ray (SXR) radiation ([0.1 keV; 20 keV]) of magnetic fusion plasmas is a standard way of accessing valuable information on particle transport and MHD. Generally, like at Tore Supra (TS), the analysis is performed with a 2D tomographic system composed of several cameras equipped with Silicon Barrier Diodes (SBD). On WEST the installation of an upper divertor masks many of the actual TS vertical diodes so that no proper tomography is possible. This paper presents the design of a new SXR diagnostic developed for the WEST project, based on a triple Gas Electron Multiplier (GEM) detector [1]. This detector works in photon counting mode and presents energy discrimination capabilities. The SXR system is comprised of two 1D cameras (vertical and horizontal views respectively), located in the same poloidal cross-section to allow for tomographic reconstruction. An array (10 cm x 2 cm) consists of up to 128 detectors in front of a Beryllium pinhole (equipped with a 0.1 mm diameter diaphragm) inserted at about 60 cm depth inside a cooled thimble in order to retrieve a wide plasma view. Acquisition of low energy spectrum is insured by a helium buffer installed between the pinhole and the detector. Complementary cooling systems (water and alcohol) are used to maintain a constant temperature (25°C) inside the thimble. Preliminary simulations performed to size and position the detector and its electronics inside the vertical thimble are presented, in particular estimation of photon fluxes using SANCO code [2], magnetic field and temperature variation using ANSYS [3] affecting GEM spatial resolution and signal quality. As a conclusion, perspectives about tomographic capabilities of the new system for studying impurity transport are given. [1] F. Sauli, Nucl. Instr. and Meth. in Phys. Res. Sect. A, vol. 386, no. 2, 1997. [2] L. Lauro Taroni et al., Proc. 21st EPS Conference, Montpellier,France, Vol. I, p. 102 (1994). [3] C. Portafaix, Proc. e 26th SOFT conference, Porto, 2016

Id 265

Abstract Final Nr. P1.053

Development of the ITER Continuous External Rogowski: from conceptual design to final design

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Magnetic measurements provide one of the most important diagnostics in tokamaks as they essentially contribute to safety, machine protection, plasma control, and plasma physics analysis. In ITER, an accurate measurement of plasma current, with high reliability, is mandatory as this parameter is used to demonstrate licensing compliance with regulatory limits. For that purpose, several independent measurements based on magnetic diagnostics have been proposed. Rogowski coils are standard inductive sensors for current measurement in many applications. In ITER, 3 Continuous External Rogowski (CER) coils are to be installed in the casing of the Toroidal Field Coils (TFC). These sensors are remarkable from several points of view: overall length is about 36 m, designed as a double helix to improve its sensitivity, located in the TFC casing at 4.5 Kelvin. The design and development of CER sensors has begun in 2005. As a first step, several sensor design options were analyzed for compliance with ITER requirements in terms of measurement accuracy and integration inside the TF casing. In the next step, prototypes of a selected CER design were built and tested. For that purpose, an extensive R&D program (10 test beds, 100 tests) has been performed. Finally the CER integration, assembly procedure and factory/site acceptance tests have been defined according to the TF manufacturing and assembly constraints. Each design phase (conceptual, preliminary and final) has been validated through a formal design review. This paper deals with the detailed description of the overall development cycle of the CER sensors. It discusses the design, the test and the integration of the sensor in the TFC casing. Feedback on the issues met during each development phase will be presented.

Id 306

Abstract Final Nr. P1.054

The new calorimetric diagnostic of WEST and its applications

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Since 2001, Tore Supra Primary Heat Transfer System is equipped with a calorimetric diagnostic. It was successfully used to measure the energy repartition on the Plasma Facing Components the power balance of the Tokamak and to check the proper operation of the various cooling circuits. In 2002, it was also used for the alignment of the LPT sectors and to derive the correlations between the Ripple Losses, the injected Lower Hybrid Heating powers and plasma parameters. It allowed to measure the averaged energy on a PFC and it is a complement of the IR viewing system which measures the surface temperatures and the heat flux repartition on a PFC. Reverse thermal methods were developed to compute the average flux impacting every single PFC. The thermal transfer function from every PFC must be derived. They have been experimentally measured during long shots In WEST, Tore Supra will be equipped with a tungsten divertor and tungsten coated Plasma Facing Components, in order to test the ITER divertor tungsten monoblocks in ITER relevant flux conditions during long pulses. The Primary Heat Transfer System circuits will be adapted to the new PFCs configuration. The calorimetric diagnosis will have to be adapted to the new circuit. For WEST configuration, we will use the same principle and sensors lay-out as the one previously used on Tore Supra. It will be possible to measure the thermal energy from 30° toroidal extension sectors corresponding, in the case of the lower part divertor, to 38 PFCs, containing each of them 35 tungsten monoblocks. It is planned to set-up temperature sensors in the vacuum vessel at the inlet and outlet of some lower divertor PFCs in order to measure the energy on this key component. Another improvement will be real-time derivation of the power on the PFCs from the calorimetric measurements. As the first pulses of WEST will be short, we will compute for each of them their thermal transfer functions with ANSYS/Fluent. In WEST, the data from the calorimetry will also be useful for the calibration of the IR viewing system, as multiple reflexions will pertubate its measurement. The paper will present the specifications and lay-out of the calorimetric diagnostic for WEST and the methods which are being developed for the derivation of the incident heat power average on all the PFCs of WEST.

Id 503

Abstract Final Nr. P1.055

Mechanical design and thermo-hydraulic simulation of the infrared thermography diagnostic of the WEST tokamak

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The WEST (Tungsten (W) Environment in Steady state Tokamak) project is a partial rebuild of the Tore Supra tokamak to make it an X-point metallic environment machine aimed at testing ITER key technologies in relevant plasma environment. For the safe operation of the WEST tokamak, Infra-Red (IR) thermography is a crucial diagnostic as it is a sound and reliable way to detect hotspots or abnormal heating patterns on the plasma facing components (PFCs). Thus WEST will be fitted with eleven middle/short-IR (1.5-2µm or 3-5µm) cameras in the upper port plugs to get a full view of the critical PFCs (in particular the new lower divertor) and heating RF antennas and one camera at the equatorial level to monitor the new upper divertor and the first wall. The change in the layout of the plasma chamber yields a partial obstruction of the upper port plug which imposes a radial shift in the position of the existing upper IR diagnostic to avoid making them blind. In turn, this radial shift leads to a low-grazing angle on the heating antennas which degrade the optical resolution and make metallic reflections a potential show-stopper. Thus the position shift and the change of plasma shape imply the design of a new actively cooled optical head, the use of an alternative solution to overcome performance and integration issues on the antenna view and a re-evaluation of the incident thermal loads such as plasma radiation and fast electron ripple losses. This paper describes the design of the up-to-date optical systems along with the hydraulic analysis and the thermal and mechanical finite element analysis conducted to ensure adequate heat extraction capabilities. Boundary conditions and simulation results will be presented and discussed as well as technological solutions retained.

Id 510

Abstract Final Nr. P1.056

The ITER Equatorial Visible/Infra-Red Wide Angle Viewing System : status of design and R&D

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The Equatorial Visible/Infra-Red Wide Angle Viewing System (WAVS) is one of the ITER key diagnostics owing to its role in machine investmentprotection through the monitoring of plasma facing components by Infra-Red thermography and visible imaging. Foreseen to be installed in 4 equatorial port plugs to maximize the coverage of divertor, first wall, heating antennas and upper strike zone, the WAVS will likely be composedof 15 lines of sight and 15 optical systems transferring the light along several meters from the plasma through the port plug and interspace up to detectors located in the port cell. After a conceptual design phase led by ITER Organization, the design is being further developed through a Framework Partnership Agreement signed between the European Domestic Agency, Fusion for Energy, and a consortium gathering CEA, CIEMAT and Bertin. The next step, the System Level Design phase, will enable to consolidate the WAVS specifications as well as the performance realistically achievable (taking into account respective ITER and project constraints). This phase will be preceded by a preparatory phase aiming at clarifying the WAVS functionalities and identifying critical prototyping and testing. The outcomes of this preparatory phase are reportedin this paper. First the resulting updated measurement specifications assigned to WAVS are presented allowing for a clear separation of measurement parameters mandatory for machine protection (with stringent requirements) from those relevant for machine control and physics studies. Secondly activities performed to ensure sound foundations for the WAVS design are summarized, such as the establishment of requirements, definition of interfaces, functional analysis and identification of urgent R&D tasks. Among the latter, particular attention will be paidto first mirror issues (erosion/deposition, active cooling, shutter…), irradiation tests of optical components and prototyping of the optical alignment compensation system to cope with differential movements between port plug and interspace.

Id 833

Abstract Final Nr. P1.057

Feature selection for disruption prediction from scratch in JET by using genetic algorithms and probabilistic predictors

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Recently, a probabilistic classifier has been developed at JET to be used as predictor from scratch. It has been applied to a database of 1237 JET ITER-like wall discharges (of which 201 disrupted) with good results: success rate of 94% and false alarm rate of 4.21%. A combinatorial analysis between 14 features to ensure the selection of the best ones to achieve good enough results in terms of success rate and false alarm rate was performed. All possible combinations with a number of features between 2 and 7 were tested and 9893 different predictors were analyzed. An important drawback in this analysis was the time required to compute the results that can be estimated in 1731 hours (~2.4 months). Genetic algorithms (GA) are search algorithms that work simulating the process of natural selection. In this article, the GA and the Venn predictors are combined with the objective not only of finding good enough features within the 14 available ones but also of reducing the computational time requirements. Four different performance metrics as measures of the GA fitness function have been evaluated. The measure F1-score needed 15 generations to reach the highest fitness value, equivalent to assess 420 predictors at 73.5 hours. Accuracy-rate measure required 12 generations (336 predictors at 58.8 hours). Matthew's correlation coefficient (MCC) found the most relevant features after 8 generations (224 predictors, 39.2 hours), and the best assessment was the measure called Informedness (the difference of success rate and false alarms), with just 6 generations (168 predictors at 29.4 hours). In all cases, the results show a success rate of 94% and a missed alarm rate of 4.21%.

Id 394

Abstract Final Nr. P1.058

Data archiving system implementa2on in ITER's CODAC Core System

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The aim of this work is to present the implementation of data archiving in ITER's CODAC Core System software. This first approach provides a client side API and server side software allowing the creation of a simplified version of ITERDB [1] data archiving software, and implements all required elements to complete data archiving flow from data acquisition until its persistent storage technology. The client side includes all necessary components that run on devices that acquire or produce data, distributing and streaming to configure remote archiving servers. The server side comprises an archiving service that stores into HDF5 [2] files all received data. The archiving solution aims at storing data coming for the data acquisition system, the conventional control and also processed/simulated data. The work includes the presentation of the complete architecture with the most relevant technical details, and its applicability on implemented integration cases: simplified demonstration with a data acquired using NI PXI-6259 I/O cards, a camera acquisition system using NI FlexRIO [3,4] and a generic EPICS-based Nominal Device Support. [1] Gheni Abla, Gerd Heber, David P. Schissel, Dana Robinson, Lana Abadie, Anders Wallander, Sean M. Flanagan, "ITERDB—The Data Archiving System for ITER", Fusion Engineering and Design, Available online 4 March 2014, ISSN 0920-3796 [2] "HDF-5 home web page", http://www.hdfgroup.org/HDF5/ [3] Sanz, D.; Ruiz, M.; Castro, R.; Vega, J.; Lopez, J.M.; Barrera, E.; Utzel, N.; Makijarvi, P., 'Implementation of Intelligent Data Acquisition Systems for Fusion Experiments Using EPICS and FlexRIO Technology,' Nuclear Science, IEEE Transactions on , vol.60, no.5, pp.3446,3453, Oct. 2013 [4] E. Barrera, M. Ruiz, D. Sanz, J. Vega, et al. , Test bed for real-time image acquisition and processing systems based on FlexRIO, CameraLink, and EPICS, Fusion Engineering and Design, Available online 4 March 2014, ISSN 0920-3796

Id 594

Abstract Final Nr. P1.059

Control and Data Acquisition for dual HIBP diagnostics in the TJ-II stellarator

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The dual 150 keV Heavy Ion Beam Probe diagnostic (HIBP2) was recently installed in the TJ-II stellarator, it consist of two subparts called HIBP1 and HIBP2, operating simultaneously ¼ of torus apart from each other at similar sectors of TJ-II. A new control and data acquisition system (C&DAS) of high performance and high reliability is in operation for dual HIBP. The 20-channel stainless steel split plate is used as ion beam detector in HIBP2, while 8-channel detector is in operation in HIBP1 at the moment. A new specific front end electronics has been designed. It is attached directly to the detector in order to minimize disturbance of the weak (0.1-10 nA) analog signal. For this reason, the front end electronics has been designed to work in high vacuum into the analyzer chamber. This board is a low noise, high frequency twenty channels current to voltage converter amplifier for use with current sources. It has a 1 Mhz bandwith and 1.1x107 V/A gain. Specially designed 500 kHz bandwidth optocouplers isolate the 50 kV energy analyzer from the acquisition boards. The HIBP C&DAS is based on the National Instruments PXI platform. PXI is a high-performance platform for measurement and automation systems. PXI units make analog and digital data acquisition and control of the dual diagnostic: manages trigger signals, controls electrostatic plates to guide beams, digitalizes detector signals, controls and monitorizes all diagnostic parts like ion injector, beam focusing and safety protections. Finally, it saves data into the TJ-II database. PXI chassis is monitored from an operator’s PC in the TJ-II control room, that is used for injector, analyzer and beam control High Voltage and current settings, and the vacuum valve controller. Both PXI and supervisor PC are running LabVIEW programs. The work is supported by RFBR grant 14-02-01182 and STCU P-507.

Id 707

Abstract Final Nr. P1.060

Dual Heavy Ion Beam Probing in the TJ-II stellarator

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A new set of Heavy Ion Beam Probe diagnostic (HIBP2) was recently installed in TJ-II flexible heliac (<a>=0.22 m, <R>=1.5 m, Bt=0.95 T, PECRH =0.6 MW, PNBI =1 MW) 1/4 of the torus apart from the routinely operating HIBP1. Both systems have 150 keV Cs+ accelerators with ion current up to 150 µA and mult ichannel energy analyzers with similar beam detectors and front-end electronics with a bandwidth up to 500 kHz and 2 MSps digitizers. Front-end electronics consists of the preamplifiers and optocouplers with optimized flatness in the gain directed to the coherence analysis. Simultaneous operation of both HIBP1 and HIBP2 has a purpose to study the long-term toroidal correlation of the oscillations of the core plasma parameters: electrostatic potential, plasma density and poloidal magnetic field, and is directed to the study of Zonal Flows in toroidal plasmas. The initial simultaneous operation of both HIBP1 and HIBP2 in low-density ECRH plasma and high-density NBI plasma shows the reliability of the dual HIBP system. The first results shows the features of the long range correlations for various types of the global modes in TJ-II plasmas. The work is a result of the long term trilateral cooperation between participating institutes. It is supported by RFBR grant 14-02-01182 and STCU P-507 and the Spanish National Research Plan ref ENE2012-38620-C02-01.

Id 802

Abstract Final Nr. P1.061

Boiling bubbles monitoring for the protection of the LIPAc beam-dump

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LIPAc is a prototype of one of the two IFMIF accelerators. It will not have a target and hence a dump is needed to stop the 125 mA beam of 9 MeV deuterons. It will be cooled with water in order to remove the 1.1 MW heat power deposited by the beam. The parameters of the cooling loop have been selected in such a way that no boiling is expected during the normal operational range. If the beam conditions change outside allowable margins (for instance, too focused beam or too misaligned beam), the beam dump material in contact with the water reaches locally temperatures higher than the water saturation point. Thus, local boiling happens, being a good indication of abnormal conditions and can be used to trigger the beam shutdown through the Machine Protection System before any mechanical damage occurs. A prompt detection of this situation can be done thanks to the use of hydrophones recording the boiling sound. The sensors must have a broad and flat frequency range as well as a high sensitivity to monitor the sound produced in different situations. Miniaturization is necessary due to space restrictions. In order to study the characteristics of the bubbles produced in different situations, an experiment has been carried out on a 1:1 prototype of the beam dump. A hydrophone was placed inside the cooling loop and the sounds produced by the bubbles were recorded to analyse their main properties. The heating was provided by means of an isolated "hot finger", 500 W/cm2 of controlled localized power density. The experimental setup has allowed a good knowledge of the features of noises produced by localized boiling. In particular, a full characterization of the time-spectral content has been done for different cooling flows, allowing the design of the final protection subsystem.

Id 1003

Abstract Final Nr. P1.062

The upgraded Collective Thomson Scattering diagnostic of FTU

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The 140 GHz Collective Thomson Scattering (CTS) diagnostic installed on the Frascati Tokamak Upgrade (FTU) has been upgraded. The new system now allows to detect the thermal CTS signal (for the first time with the probe gyrotron frequency below the first harmonic electron cyclotron resonance) and to study the impact of the parametric decay instability (PDI) process on the launched millimeter-wave beam as well as on the quality of the scattered signals with collective nature. In particular, the second issue must be studied in detail to investigate if some operational limits exist for the use of the CTS diagnostic in ITER. The electron cyclotron front-steering antenna and transmission system have been complemented with a new receiving line for the detection of the scattered signal. It consists of circular corrugated waveguide and a set of 5 mirror boxes, to match the existing quasi-optical receiving line feeding the heterodyne radiometric receiver. The antenna configuration is such that the probe beam and the detected signal directions are poloidally symmetric and it allows to vary the position of the scattering volume either on the equatorial plane or out of it, as well as to change the scattering angle with fast angular movements of the mirrors. These movements can be performed during the plasma shot, scanning a wide angular range in both toroidal and poloidal directions with the probe and with the receiving mirror. The data acquisition and analysis system is improved with the addition of a new fast digitizer, allowing the reconstruction of the scattered radiation spectra by direct sampling and Fourier transform of the down-converted signal. This ensures the suitability of the new diagnostic either to carry out thermal CTS measurements or to characterize the anomalous signals presently ascribed to PDI, observed in presence of MHD phenomena.

Id 209

Abstract Final Nr. P1.063

Requirements for tokamak remote operation: application to JT-60SA

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Remote operation and data analysis are becoming keys requirement of any fusion devices. In this framework a well conceived data management system integrated with a suite of analysis and remote tools are essential components for an efficient remote exploitation of any fusion device. The following components must be considered: data archiving data model architecture; remote data and computers access; pulse schedule, data analysis software and remote tools; remote control room specifications and security issues. Mandatory to minimize resources demands and learning time, solutions should be shared with operating and in construction fusion experiments wherever possible. This applies in particular to the data system where some solutions are emerging as de-facto standards: like for example the HDF5 data format and the MDSplus data acquisition and data access package. The definition of a device-generic Physics Data Model and a signal naming convention play also important roles in improving the ability to share solution and reducing learning time. As for the remote control room, the implementation of an Operation Request Gateway has been identify as an answer to security issues meanwhile remotely proving all the required features to effectively operate a device. Previous requirement have been analyzed for the new JT-60SA tokamak device. Remote exploitation is paramount in the JT-60SA case which is expected to be jointly operated between Japan and Europe. Due to the geographical distance of the two parties an optimal remote operation and remote data-analysis is considered as a key requirement in the development of JT-60SA. It this case the effects of network speed and delay have been also evaluated and tests have confirmed that the performance of various connection and data access methods can vary significantly depending on the technology used. For this reason a preliminary selection of connection technologies and methods has been carried out.

Id 718

Abstract Final Nr. P1.064

3D, LTCC-type, High-Frequency Magnetic Sensors for the TCV Tokamak

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Innovative high-frequency magnetic sensors are being designed and manufactured in-house for installation on the TCV tokamak. These sensors are based on combining the Low Temperature Co-fired Ceramic (LTCC) and the classical thick-film technologies, and are to provide measurements in the frequency range from 5kHz to 1MHz of the 3D perturbation to the toroidal (deltaBTOR), poloidal (deltaBPOL) and radial (deltaBRAD) magnetic fields. The main design principles are aimed at increasing the effective area (NA_EFF) and self-resonant frequency omega0=1/(L_SELF*C_SELF)1/2 of the sensor in each of the three measurement axes, while reducing the parasitic capacitance between them. An optimization algorithm for L_SELF and C_SELF is used, based on that developed for 1D LTCC sensors [1]. The main design constraints are set by the measurements and installation requirements. The physics requirements are those for the study of fast ions and MHD physics due to the forthcoming installation of a high-power / high-energy NBI system on TCV. Thus 4 to 7 sensors will be installed, at close-by locations along the toroidal coordinate at the same poloidal location, to measure toroidal mode numbers up to n=40. The 3D measurement axes have to be centred at the same position on each sensor. Regarding the installation requirements, the planar size of the sensor cannot exceed 7x7cm, while its depth must remain under 1cm. Thus, the deltaBRAD measurement is obtained through a single planar coil on the alumina base substrate, whereas the deltaBTOR and deltaBPOL measurements are obtained by arraying in series a number of identical multilayer LTCC sub-assemblies with coils perpendicular to the base. [1] D.Testa et al., Fusion Science and Technology 59 (2011), 376-396.

Id 645

Abstract Final Nr. P1.065

Advanced Remote Operation of the GOLEM Tokamak

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Tokamak GOLEM at the Czech Technical University, Prague serves as an educational device making tokamak operation accessible to students worldwide via a web application. The web application has two levels: -Basic level of remote operation enables students to set up basic discharge parameters, necessary to create plasma: toroidal magnetic and electric field, working gas pressure and tools for pre-ionization of the working gas. Consequently, measured data from basic diagnostics are acquired and presented in a hypertext form [1]. More than 800 remote discharges from foreign sites have been already successfully performed in the frame of FUSENET (Fusenet Education Network). -The second level of operation allows to affect the vertical position of plasma using the horizontal magnetic field. It operates in two modes: * pre-programmed control of horizontal magnetic field scenario, * LabVIEW based real-time system, which controls the horizontal magnetic field in response to the currently measured vertical position of plasma. The system now enables remote participants to study the effect of horizontal magnetic field on the vertical plasma position using their own approach. Relative plasma-life duration prolongation of more than 20% is now possible with respect to the mode without the stabilization. [1] V. Svoboda, B. Huang, J. Mlynar, G.I. Pokol, J. Stockel, and G Vondrasek.Multimode Remote Participation on the GOLEM Tokamak. Fusion Engineering and Design, 86(6-8):1310–1314, 2011

Id 739

Abstract Final Nr. P1.066

The effect of the accuracy of toroidal field measurements on spatial consistency of kinetic profiles at JET

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Some kinetic profile measurements rely on magnetic equilibrium reconstruction and thus precise measurement of the toroidal magnetic field for spatial mapping. At the JET tokamak spatial inconsistencies up to 3-5cm are found between electron density measurements (reflectrometry, high resolution (HRTS) and LIDAR Thomson scattering, Lithium Beam Emission Spectroscopy, specific interferometry lines of sights) and electron temperature measurements (HRTS, electron cyclotron emission). Historically this has been associated with an error in the toroidal field measurement and toroidal magnetic field corrections are commonly applied to resolve these spatial inconsistencies. Recently, in December 2013, a high precision (up to 0.1%) optical fibre current measurement has been installed on a toroidal field coil on JET [1], giving a unique opportunity to precisely measure the toroidal field current and check whether the spatial inconsistency is in fact related to the measurement of the toroidal field. This paper compares the historical toroidal field measurements (Hall probe, shunt and Rogowski coils) with each other and the new optical fibre measurement showing an error up to 1.6% between the Rogowski coil pair used in the magnetic equilibrium reconstructions and the high precision optical measurement. This explains partially the spatial inconsistency of some kinetic profile measurements, however not all. Thus new magnetic equilibrium reconstructions based on the high precision optical measurement and statistical model selection methods [2] are used to explore further possible candidates responsible for the remaining spatial inconsistency of the kinetic profile measurements. [1] See poster by R. Salmon presented at the same conference. [2] O.J.W.F. Kardaun, Classical Methods of Statistics, Springer This work was supported by EURATOM and carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Id 403

Abstract Final Nr. P1.067

Thermal analysis for optimization of the optical duct of the ITER core CXRS diagnostics

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The First Mirror (M1) as a part of the ITER cCXRS diagnostics is responsible for acquisition and transportation of the optical signal from the plasma to the corresponding spectrometer. The M1 is the most vulnerable component of this system working in severe conditions caused by its location in the direct view of the plasma. Since the M1 is located in the near vicinity of a diagnostic first wall (DFW) surface where the level of thermal loads reaches 500 kW/m2 (by particles and X-rays/UV radiation) the necessity arises in its thermal protection. A cooling of the mirror should stabilize the temperature level of its surface and utilize the absorbed thermal flux, while passive protection measures are intended to reduce the thermal flux on the mirror by its positioning at some depth in a DFW block. This embedding of the M1 into the DFW forms a cylindrical duct (called as an “optical channel”) intended to reduce the mirror exposure angle. The authors report on the results of numerical study of radiation fluxes going through a baffle-like wall duct aimed to decrease a part of energy reaching M1. The authors investigated the processes of interaction between radiation fluxes from the plasma in different parts of spectrum, the optical channel walls and the mirror surface. The investigation was performed with 3D thermal numerical models taking into account not only the optical characteristics (varied emissivity) of the optical channel but also its temperature state. Based on the numerical analysis an optimized arrangement of the baffles was found for the optical channel. It was shown that such measures like the deeper M1 positioning in the shielding, duct configuration and baffles implementation made it possible to reduce heat loads on the M1 by factor of 2000-5000.

Id 550

Abstract Final Nr. P1.068

Advanced Methods for Image Registration Applied to JET Videos

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The last years have witnessed on JET a significant increase in the use of digital cameras, which are routinely applied for imaging in the IR and visible spectral regions. One of the main technical difficulties, in interpreting the data of camera based diagnostics, is the presence of movements of the field of view. Small movements occur due to machine shaking during normal pulses while large ones may arise during disruptions. Some cameras show a correlation of image movement with change of magnetic field strength. For deriving unaltered information from the videos and for allowing correct interpretation, an image registration method, based on highly distinctive Scale Invariant Feature Transform (SIFT) descriptors and on the Coherent Point Drift (CPD) points set registration technique, has been developed. The method ensures a reliable determination of the alignment parameters and their corresponding uncertainty. The algorithm incorporates a complex procedure for rejecting outliers. The method has been applied for vibrations correction to a representative set of videos collected by the JET wide angle infrared camera. It proved to provide reliable image vibration correction when the camera is operated in full view. Another application has been developed for the JET fast visible camera, where the image rotation artefacts occur due to the sensitivity of the image intensifier to the magnetic field (above a certain threshold). The method has proved to be able to deal with the images provided by this camera frequently characterized by low contrast and a high level of blurring and noise. It has also the advantage of using directly the input images as they are provided by the cameras without any pre-processing step. The parameters of the algorithm are the same for all videos in the database.

Id 52

Abstract Final Nr. P1.070

Calculating the 3D magnetic field of ITER for European TBM studies

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In this contribution we describe the calculation of unprecedentedly detailed ITER 3D magnetic fields, with emphasis on ferromagnetic components. The fields are mainly intended for studies of the effect of the European test blanket module (TBM) to fast ion confinement, as a local leak in confinement could lead to plasma facing component damage. We model detailed ferritic material properties and magnetic component geometry (toroidal and poloidal field coils, TBMs, ferritic inserts). The main tool is the commercial COMSOL Multiphysics finite element method (FEM) platform. We describe the process that starts from imported ITER CAD models and ends in exporting the magnetic field and the vector potential to fast ion and 3D equilibrium codes. The ferromagnetic geometry and coils are appropriately simplified and the corresponding change in the ferromagnetic mass is compensated for. The electric currents from coils and the plasma current are included. The sensitivity to the boundary conditions of zero vector potential far away from the machine is checked. No FEM calculation can match the precision of a direct Biot-Savart law integration for magnetic fields from known coil geometry. Therefore we only use from COMSOL the field due to the magnetization, which is extracted by a secondary magnetostatic calculation that uses the electromagnetic calculation as input. We also describe the exporting of the fields in high quality using a carefully constructed FEM mesh, which facilitates the use the fields unsmoothed. As final results we show ripple maps and illustrations of Fourier decompositions of the resulting fields. The results are compatible with other similar calculations, but include a more detailed perturbation field. The FIs produce a strong Fourier mode at n=18 as expected. The TBMs produce a wider spectrum perturbation.

Id 79

Abstract Final Nr. P1.071

Safety Analysis for CFETR superconducting magnet system

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CFETR superconducting tokamak is a national scientific research project of China. The plasma major radius R is 5.7 m and minor radius r is 1.6 m. The plasma center magnetic field strength (at R=5.7 m) is 5.0 T. The main goal of the project is to build a fusion engineering Tokamak reactor with its fusion power is 50~200 MW and should be self-sufficiency by blanket. The magnet system including TF, PF, CS and CC magnets has been detailed designed according physical parameters. All the coils are designed with superconducting materials and cooled by force flow Helium. The stored magnetic energy of CFETR magnet system will be large, for example, the stored energy of TF magnet system is about 36 GJ. The total current for each TF coil will be 8.9 MA, while for CS coil will be about 15.8 MA- 23.7 MA. Thus, according to the safety assessment criteria of large superconducting magnet, all the CFETR magnets should be steady operation during the normal operation, even in the extreme conditions, such as mechanical instability, quench, peak nuclear heating caused by fast neutron flux and so on. In this paper, the main safety assessment work for CFETR superconducting magnet will include three categories. First, superconductor stablity and quech protection analysis results were discussed according to the operation parameters. Second, the structure safety analysis work was detailed carried out in the event of plasma discharge and quench accident. Third, local hot spot problem arising from fast neutron flux and quench accident were calculated. All these calculation and analysis work should take consideration and be useful to examine the magnet safety before the R&D stage.

Id 127

Abstract Final Nr. P1.072

Design of the CFETR CS model coil

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Institute of Plasma Physics Chinese Academy of Sciences has been designing and developing the Central Solenoid (CS) model coil for China Fusion Engineering Test Reactor (CFETR). The goal of the CFETR CS model coil program is to validate and consolidate the design and manufacturing feasibility. The CFETR CS model coil consisting of inner and outer modules, is designed to generate 13 T of the rated magnetic field with a 45 kA operating current. The inner module which has an inner diameter of 1.1 m, an outer diameter of 1.96 m, a height of 2.0 m, use the ITER CS conductor. The outer module which has an inner diameter of 1.98 m, an outer diameter of 2.9 m, a height of 2.39 m, use the ITER PF6 conductor. The use of NbTi conductor for the outer module is a compromise between the need of 13T rated magnetic field and the limited budget. The CS model coil will produce a maximum field of 13 T in the Nb3Sn inner module, and a maximum field of 6.4 T in the NbTi outer module.

Id 227

Abstract Final Nr. P1.073

Thermal-hydraulic analysis of the Zig-Zag type HTS-CL heat exchanger

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The heat exchanger (HEX) of the High Temperature Superconducting Current Lead (HTS-CL) connects the superconducting section and the water-cooled cables, the temperature of the HEX ranges from about 50K to 300K; therefore active cooling is used to remove the high ohmic heat. The Zig-Zag type design has high convection heat transfer coefficient and large wet area, but the pressure drop is also high; accurate CFD analysis is required to predict the temperature distribution and pressure drop of the HEX. This paper describes the thermal-hydraulic analysis of the HEX and takes the ITER 68kA HTS-CL prototype as an example. Two geometry modeling methods are compared, one is called 'unit' method, which builds only a unit section of the HEX, while the 'global' method uses the whole HEX to increase accuracy. The turbulent model choice decides the calculation accuracy and resource cost, two equation k-epsilon and k-omega model are compared to one equation S-A model. The boundary condition of the HEX is a critical issue in the analysis, especially at the room temperature end, no active control is applied, therefore it's difficult to set a proper boundary, different solutions are also compared. According to the efforts, the HEX thermal-hydraulic analysis procedure can be concluded and applied in future applications to save time and increase the confidence in results.

Id 275

Abstract Final Nr. P1.074

Design and Analysis of the New Current Feeders for EAST Device

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In the Experimental Advanced Superconducting Tokamak (EAST)device, the current feeder system which connects the SC magnet system and the power system consists of 13 pairs of bus-lines. The current feeder system of EAST device has been operating for eight years, with the improvement of experimental parameters and increasein diagnostic devices, the two current lead tanks are going to be moved to the new power hall which is 30 meters away from the EAST device. And as a result, we need to redesign and fabrication the current feeder system. In this paper, the design and analysis of electromagnetic force constraint, structural support, electrical insulation, thermal insulation and Vacuum barrier for the new EAST current feeder system are introduced. The results verify the reasonability and reliability of the design of the new EAST current feeder system.

Id 381

Abstract Final Nr. P1.075

Design, fabrication and electromagnetic analysis of EAST VS coils

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EAST is the first full superconducting tokamak device in the world. Due to the elongated device, the plasma is unstable to a vertical displacement. Its control relies on the active feedback control coils [1, 2] and the passive feedback coils [3]. The active feedback control coils, often called vertical stabilization (VS) coils, are located inside vacuum vessel and besides the outside edge of the plasma [4, 5]. After six years running, the insulation material of the original VS coils is aged and is hard to operate steadily. Besides, the ceramic rings are damaged because the coils are suffered by the huge shock electromagnetic force at the feeder part. Therefore, the coils should be technical renovation and transformation. This report is mainly about the design parameter, electromagnetic and water-cooling analysis and fabrication process. As the feeder suffers huge vertical shock force, conventional organic insulating material cannot meet the requirements. We use magnesium oxide as insulating material and add bellows to the ends of the feeders. Water-cooling calculations show that the temperature of copper conductor and water is linearly related to the distance of the point and the water inlet. In order to assure the temperature of outlet water not be heated to boil, the relation of the minimum velocity of water and operation ratio is about v=6K when the current in the coil is 20kA.

Id 448

Abstract Final Nr. P1.076

Preliminary Study on Advanced Technology for Fusion Power Supply System

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Owing to the significant progress on fusion research and the energy resource issue over the world in recent years, some ambitious research plans for fusion project, such as Demo reactor and even fusion power plant have been in program. The system scale and technical requirements are getting much higher and higher for the power system. Review on the present technology for high power conversion, the so called Thyristor-based technology is the dominant technology for long time. But its inherent disadvantages are apparent and serious. However, the advent of high rating full controlled semi-conductor devices with advanced power electronics technology have brought a new alternative into industrial application in recent ten years. And its rapid development and great progress make it not only possible but also feasible to power supply system in fusion technology nowadays. It may be called Advanced Technology for Fusion Power System -'AT-FPS' Its excellent characteristics and attractive features, Its feasibility, flexibility and reliability will be introduced. The application of advanced power electronics technology in fusion power supply system, such as high power conversion, intermediate energy storage, system optimization and control will be investigated. A preliminary system comparison between the two kinds of technology, taking ITER-like Tokamak as an example, including roughly cost evaluation will be present. The 'AT-FPS' will be proven as the dominant technology sooner or later in fusion Technology, the tendency is obviously without doubt.

Id 797

Abstract Final Nr. P1.077

3D ANSYS Model of an Unmitigated Quench in ITER coils

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The basic process of a quench is the conversion of stored electromagnetic energy into heat when a superconducting magnet is operated above its critical parameters. Although this is quite a common occurrence due to the relatively low energy input needed to trigger a quench, several methods can be implemented to mitigate the effect of such disturbances. However, proving that, even in the extremely unlikely event that the coil detection and protection systems failed, the magnets cannot endanger either the primary or secondary containments, is needed to be consistent with safety claims made as part of the ITER licensing process. The objective of this work is to discuss a comprehensive model for the analysis of fault conditions in superconducting magnets, and apply it to the latest ITER coil design. ANSYS, a well-known validated industry standard tool, in combination with its parametric design language, is being used to model and understand the consequences of an unmitigated quench. Lumped circuit elements representing the external electrical network drive a high resolution 3D Finite Element model of one of the magnet coils. Voltage, temperatures and volumes of melted material can be estimated in order to predict the likelihood of arcing. Several types of arc can be formed – vapour, Paschen breakdown and dielectric breakdown – depending upon the fault event, requiring integration of a model for each type, including its thermo-physical effects. The accident will continue to progresses until the inductive magnetic energy stored in the system is dissipated as Joule and local arc heating. The paper describes the details of the computational procedure and applies it to a representative magnet model.

Id 247

Abstract Final Nr. P1.078

Implications of TF coil stress limits on power plant design using PROCESS

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Systems codes capture interactions between systems in a fusion plant and trade-off parameters in order to produce a self-consistent plant design. DEMO studies using systems codes allow the optimisation of designs to maximise/minimise some figure of merit: fusion power gain or cost of electricity, for example. This paper describes the toroidal field coil (TFC) stress model in the systems code PROCESS. The TFC structure is critical in determining the plant design as it strongly influences key parameters, in particular radial build and available toroidal field. The model was compared with finite element calculations and used to investigate how TF stress impacts DEMO design; both pulsed (DEMO1) and steady-state (DEMO2) devices. A scan of allowable stress from 400-700 MPa was run for both DEMO1 and DEMO2; this is the allowable stress range for steels from the ITER TFC design. The scan (whilst minimising major radius, R0) produced a variation in R0 of 14% of the 9.25m DEMO reference, for fixed aspect ratio. R0 is a large driver of capital cost; which varied by $1bn (1990 $) over the same range. When approaching the upper limit for allowable stress other factors start limiting the plant; such as the heat load to the divertor approaching unacceptable levels (20MW/m) due to decreasing R0. These trade-offs are integral to finding suitable designs and understanding how some parameters limit the design is essential for exploring new DEMO concepts and identifying areas of future research. This paper details the recent update to the PROCESS stress model, comparisons with FEM calculations and how TF stress impacts DEMO design. Parameters with a TF stress dependency are described and their relative inuence on producing feasible solutions examined. This work was part-funded by the RCUK Energy Programme and by the European Union's Horizon 2020 programme

Id 580

Abstract Final Nr. P1.079

Thirty Year Operational Experience of the JET Flywheel Generators

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The JET Pulsed Power Supply System employs two Flywheel Generator Converters which can each supply 2600MJ to their respective dedicated magnet load coils to supplement JET's 575MW (pulsed) grid supply. In addition, one generator is also able to connect to loads in a test cell; a facility that was used in 1999 to test a high current break-switch for ITER. Since these generators entered service in 1983 they have both operated for approximately 85 000 JET pulses. Never-the-less since commissioning a number of issues have arisen, the most significant being: The liquid-rheostat used for speed control requires frequent maintenance due to the corrosive nature of the electrolyte employed. Arcing within this component was identified as being due to inconsistent dynamic fluid flow within the resistor and lead to the addition of a weir to control the electrolyte depth. Difficulties starting both machines have occurred over several periods but recently this has become more prevalent. Extensive investigations revealed that the thrust bearings were presenting excessive friction at standstill, predominantly due to the pads' surface profile. The plug braking system has suffered from two transformer failures since 2006. New transformers have been installed to replace those that were prone to failure and these have been successfully operating since 2012. Vibration in the generators is strongly correlated with component lifetime, so care is taken to monitor vibration and sound levels, and excessive levels are responded to. In 2010 an inspection took place which included EL-CID tests to identify core defects, partial-discharge measurements to monitor insulation health, as well as non-destructive testing on critical welds. In conclusion, JET's flywheels have provided effective energy storage to supplement the available grid power for this pulsed fusion experiment, and with continued maintenance and monitoring are expected to operate successfully well beyond the next decade.

Id 805

Abstract Final Nr. P1.080

Comparison of different current transducers used at JET within the range 5-100kA for plasma control and monitoring

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The JET machine operates a variety of current transducers for control and protection of the plasma, the coils and their associated power supplies. This paper reviews the various measuring technologies, within the range 5-100kA, used on JET to assist with the selection of high-current transducers for future plasma control/tokamak applications; these include Rogowski coils, Coaxial Shunts, Hall-effect transducers, zero-flux CTs and a Faraday-effect optical transducer. The paper considers cost, reliability, accuracy and usability based on up to thirty years of operational experience of the transducers. Accuracy of the magnet current measurements is important in the control of tokamak plasmas and there has been considerable effort to improve it. Recently a Faraday-effect optical current sensor has been used to measure up to 67kA in the Toroidal Field (TF) coil circuit. This measurement system has been calibrated at JET to verify it's 0.1% accuracy. In addition the data acquisition system for this measurement is automatically calibrated at the start of each JET pulse. The improved accuracy has been shown to enhance the spatial consistency of kinetic profiles at JET. Due to its portability the JET project intends to employ the same Faraday-effect current transducer to calibrate other high current transducers by temporarily fitting it to other busbars; such as those in the Ohmic Heating network.

Id 836

Abstract Final Nr. P1.081

Starting the production of the CEA JT-60SA TF coils procurement

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In the broader approach frame, the French voluntary contributor represented by CEA has awarded a contract mid 2011 for the production of 9 TF coils of the JT-60SA project to Alstom, Belfort, France. A preparatory phase was first led to set the team, product the manufacture drawings, define the manufacturing process, procure the needed tooling and prepare the quality documentation. In parallel, a qualification phase to secure the whole manufacture has proved the Alstom ability in the mastering of the needed processes to reach the requirements. The conductor bending with an accuracy better than 1 mm was demontrated. The conductor preparation has shown an insulation adhesion in a 50 MPa range even after cycling 36000 time up to 20 MPa. Joint area insulation and impregnation processes as well as semi-automatic TIG casing welding and their US check were demonstrated. One of the last qualifications was related to the winding machine operation for what a conductor length was wound and has highlighted several issues in the final geometry to solve prior starting the winding production. After careful analysis of the results and with the help of specific CEA developments on the conductor bending mechanical behaviour, an action plan to improve the winding tooling and process was defined in close collaboration between Alstom, CEA and F4E. After reviewing of the results and processes to overcome these difficulties through a PRR (Production Readiness Review), Alstom was authorized to start the winding production. A prototype double pancake was wound as first of production. In addition to the compliance with the 144 ± 3 mm pancake width all along the D shape, the straightness of the centerline in the critical straight leg part was better than 1 mm. The paper describe the completion of the last qualifications and the status of the winding production.

Id 75

Abstract Final Nr. P1.082

Quench Simulations in ITER CS Magnet with SuperMagnet Code

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During ITER plasma scenarios, the six Central Solenoid (CS) modules will be operated with fast pulsed independent currents. This peculiarity creates a real challenge for the primary quench detection system based on voltage measurements, since a small resistive signal associated with a quench has to be discriminated within the high inductive signals measured across the winding elements. As a consequence, the feasibility of a secondary quench detection based on thermo-hydraulic signals, already considered for the other ITER magnet systems, is of a particular interest for the CS magnet. Studies focus on most likely quenches, which are expected to occur at high field region (innermost turns). Simulations are performed for both “severe” and “smooth” quenches, depending on the initiated quench length and on the number of quenched conductors. For smooth quenches, the most difficult to be detected, criteria are proposed for variations of thermal-hydraulic signals which could trigger the fast discharge of the current: mass flow disturbance (at least twice higher than the expulsion effect expected during normal operation), pressure and temperature rises. Simulations are performed with the SuperMagnet code, which allows simulating in one single model the CS magnet conductors and modules, as well as the different circuits (He flow restriction at insulation breakers, isolating and quench valves) connecting the magnet to the cryoplant. Results are analyzed regarding the consequences of the quench: - on the conductors (maximal hot spot temperature), - on the connecting circuits, mainly the pressure in insulation breakers, which shall not exceed the maximum operating value guaranteed by the supplier. Further recommendations are given regarding the control of isolating valves and the opening pressure threshold of quench safety valves, so as to ensure the maximum availability and safety for the CS magnet operation.

Id 414

Abstract Final Nr. P1.083

Proposal of Experiments in HELIOS Facility for Assessment of Thermal Efficiency of ITER TF Coil Case Cooling Channels

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During ITER operation, especially in case of a plasma disruption, significant thermal loads will be generated in the stainless steel casings of the Toroidal Field (TF) coils, and the heat flux transferred to the winding pack must be limited to avoid an excessive temperature increase in the TF conductors. The temperature at the interface between conductors and casings is controlled by implementing a large number of cooling channels in the casings inner surfaces, notably in the plasma facing wall where conductors are expected to experience the most severe conditions. Therefore, the thermal efficiency of case cooling channels must be assessed, in order to ensure the safe operation of TF conductors. For this purpose, experiments under realistic cryogenic conditions are foreseen in the HELIOS test facility at CEA Grenoble. The reference design to be experimentally qualified includes a stainless steel pipe inserted in a rectangular groove, the material in-between the pipe and the case groove being a charged resin characterized by an enhanced thermal conductivity. The present study aims at identifying suitable experimental configurations. The first proposed experiment is a heat exchanger, where the inner channel is the sample to be qualified. Test conditions, such as inlet temperatures, mass flows or sample length, are investigated in order to maximize the sensitivity of the measured outlet temperatures to the efficiency of the transverse heat exchange. The second experimental configuration is a single cooling channel sample inserted between a hot heat source and a cold one. This layout allows evaluating the shielding efficiency of the cooling channel simply comparing the prescribed heat flux on hot side with the power extracted by the fluid. Both steady state and dynamic test conditions are analysed in order to gain the best characterization of the shielding efficiency in representative conditions. Finally, a preliminary test program is proposed.

Id 655

Abstract Final Nr. P1.084

Erosion of beryllium under high transient plasma heat loads

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Beryllium is considered as an armor material for the ITER first wall. Primary reasons for the selection of beryllium as an armour material for the ITER first wall are its low Z and high oxygen gettering characteristics. During plasma operation in ITER, beryllium will be suffered by low cyclic heat loads (normal events) and high transient heat loads (ELMs, disruptions, VDE, etc) which may affect on beryllium surface and joints to the heat sink. These transient loads cause rapid heating of beryllium surface and can result in some processes in surface and near-surface regions such as material loss, melting, cracking, evaporation and formation of beryllium dust as well as retention of hydrogen isotopes both in the armour and in the dust. It is expected that erosion of beryllium under transient plasma loads will determine a lifetime of the ITER first wall. This paper presents the results of recent experiments with two ITER beryllium grades: TGP-56FW and S-65C under intense transient plasma heat loads on QSPA-Be plasma gun facility. The QSPA-Be plasma gun, a quasi-stationary plasma accelerator, provides hydrogen (or deuterium) plasma heat loads in the range of 0.2-5 MJ/m2 and a pulse duration 0.5 ms corresponding to the ITER ELMs and disruptions. The Be/CuCrZr mock-ups were tested by deuterium plasma streams (5 cm in diameter) with pulse duration of 0.5 ms and heat loads of 0.6-1 MJ/m2. The angle between plasma stream direction and target surface was 30º. During the experiments the temperature of Be tiles has been maintained about 500ºC. The beryllium mock-ups were exposed to up to 100 shots. After 10, 40 and 100 shots, the beryllium mass loss/gain under erosion process were investigated as well as the evolution of surface microstructure and cracks morphology.

Id 881

Abstract Final Nr. P1.085

An Investigation of the Effectiveness of Pulsed Phase Thermography for Detection of Disbonds in HIP-Bonded Beryllium Tiles in ITER Normal Heat Flux First Wall (NHF FW) Components

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The manufacture of Normal Heat Flux First Wall (NHF FW) components for ITER brings with it a requirement for guaranteed repeatability and high quality of output. This is important considering the high costs of replacement and potential damage to underlying components resulting from failures during service. Non destructive examination (NDE) methods are therefore attractive for evaluation of NHF FW components, both prior to use and during service.Pulsed Phase Thermography (PPT) is an NDE technique, traditionally used in the Aerospace Industry for inspection of composite structures, which combines characteristics and benefits of Flash Thermography and Lock-In Thermography into a single, rapid inspection technique.The aim of this work was to evaluate the effectiveness of PPT as a means of inspection for the bond between the Beryllium (Be) tiles and the Copper alloy (CuCrZr) heatsink of the ITER NHF FW components. This is a critical area dictating the functional integrity of these components, as single tile detachment in service could result in cascade failure. PPT has advantages over existing thermography techniques using heated water which stress the component, and the non-invasive, non-contact nature presents advantages over existing ultrasonic methods. The rapid and noncontact nature of PPT also gives potential for in-service inspections as well as a quality measure for as-manufactured components. The technique has been appraised via experimental trials using ITER first wall mockups with preexisting disbonds confirmed via ultrasonic tests, partnered with Finite Element simulations to verify experimental observations. This paper will present the results of the investigation.

Id 662

Abstract Final Nr. P1.086

True surface temperature measurement on W PFC subjected to thermal fatigue loads in the FE200 facility

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Since 1991, numerous High Heat Flux (HHF) tests have been performed on various Plasma-Facing Components (PFC) and Materials (PFM) in the FE200 electron beam facility of the AREVA Technical Centre, Le Creusot (France). At the same time, non-contact temperature measurement methods have been developed and among them pyro-reflectometry enables to measure the true temperature of an opaque material without prior knowledge of its emissivity. This method is employed in Le Creusot to measure in real time the surface temperature of W PFC subjected to thermal fatigue loads up to 20 MW/m² (i.e. including surface aspect changes during the testing). Results show successful temperature measurements obtained with various W surface aspects (polished, roughened, cracked, and combined). In addition, each temperature measurement is linked to the emissivity of the surface, also determined by pyro-reflectometry. The paper describes briefly the FE200 facility, the principle of pyro-reflectometry, and results obtained with this method (temperature and emissivity measurements) during HHF tests on W PFC.

Id 161

Abstract Final Nr. P1.088

The Upgrade of EAST Divertor

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The scientific and the engineering missions of EAST are to explore the reactor related regimes with long pulse lengths and high plasma core confinement and solutions for power exhaust and particle control under steady state operation and to establish technology basis of full superconducting tokamaks. EAST can operate with double null or single null divertor mode. The goals of the divertor are to get high parameters plasma, long pulse operation and single or double null plasma shape. So the divertor geometry is designed as up-down symmetry to provide a large experimental flexibility. The divertor should have the capacity to endure required higher heat flux and provide recycling and impurity control. Up to now, three generations divertors which respectively are steel, carbon and tungsten divertor have been developed. Steel divertor is used in the commissioning test of the host machine in 2006. Carbon divertor is employed to endure thermal load of no more than 4 MW/m2 in 2008. Thus the tungsten divertor is expected to endure the heat flux up to 10MW/m2 in 2013.The carbon divertor is composed of graphite tiles, CuCrZr heat sink and stainless steel supports. And the tungsten divertor is based on cassette and mono-block technology like ITER. The tungsten divertor will be installed and employed at the middle of 2014.

Id 176

Abstract Final Nr. P1.089

Engineering conceptual design of CFETR divertor

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China fusion engieering test reactor (CFETR), is the next step fusion energy in China. Main scientific missions are: fusion power of 50-200 MW, duty cycle of 0.3-0.5 and tritium breeding ratio~1.2. The divertor, as one of the key components in CFETR, has many challenges in its design due to critical requirements including geometry accommodate with plasma configuration, cooling structure for high power exhausting, structure compatiable with remote handling, etc.. The progress of the engineering conceptual design of the CFETR divertor is presented in this conference. The CFETR divertor has been designed as a cassette structure as the support and cooling water manifold for the first wall components including inner target, two particle reflectors, dome plate and outer target. Three different geometries were designed for the first wall to accommodate with different plasma configurations. The first is for ITER-like configuration and the second for snowflake configuration. Effort was put in the design of the third one to make it is suitable for both the ITER-like and the snowflake configurations. The properties of these three geometries of the target plat were analyzed based on the parameters of expansion factor and incident angle of magnetic flux in the target in poloidal section. For different geometries of the target plat, the same cassette structure is employed to have the same interface with other components in the vacuum vessel (VV) of CFETR. The divertor has 60 modules toroidally. Each module is supported by two toroidal rails with the mechanism of pin and bearing. There are eight cooling sub-systems in the VV for the divertor, one for 45 degree sector. In the sub-system, one pair of pipes with inner diameter of 203 mm is working as the main pipes to feed and back cooling water to 7 or 8 modules which are connected parallel.

Id 281

Abstract Final Nr. P1.090

Application of E-Beam Welding in W/Cu Divertor Project for EAST

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According to the present design, the EAST W/Cu divertor consists of four principal parts: W/Cu mono-block vertical target, flat-type baffle, end-cup and flat-type dome. To realize the CuCrZr-CuCrZr joint of the W/Cu-monoblock and the baffle, the W/Cu-monoblock and the end-cup, the dome upper W/Cu/CuCrZr tile and lower CuCrZr heat sink, the electron-beam welding (EBW) technique was applied because it produced high aspect ratio fusion zones and narrow heat affected zones. In the seal welding, three issues including the microstructue of the fusion zone, the helium leak and the surface profile on the final assembled component have been investigated.

Id 901

Abstract Final Nr. P1.091

Electromagnetic and thermal analysis for blanket model of fusion reactor

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As one of the key components inner of vacuum vessel in fusion reactor, blanket face to plasma and suffer from nuclear heat coming from neutron. The major plasma disruption or the vertical displacement event all will produce toroidal eddy current in the vacuum vessel with plasma facing components, such as blanket, further produce joule heat. Plasma disruption and vertical displacement event should not merely be considered in the structural design of blankets, but also heating stresses from nuclear heat. In this paper, based on China Fusion Engineering Test Reactor (CFETR) design model, using the vector electromagnetic method, after modeling, current source loading, boundary condition setting and solving, both joule heat caused by eddy current and nuclear heat from neutron can be calculated by the ANSYS software. According to the coupling analysis results, detailed data can be presented aim to guide the blankets design of fusion reactor, insure them structure safety and more effectively to protect them from being destroyed in electromagnetic event with neutron.

Id 905

Abstract Final Nr. P1.092

Testing candidate interlayers for an enhanced water-cooled divertor target

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The design of a divertor target for DEMO remains one of the most challenging engineering tasks to be overcome on the path to fusion power. Under the European DEMO programme, a promising concept known as Thermal Break has been developed at CCFE. This concept is a variation of the ITER tungsten divertor in which the pure Copper interlayer between Copper Chrome Zirconium coolant pipe and Tungsten monoblock armour is replaced with a low thermal conductivity compliant interlayer, with the aim of reducing the thermal mismatch stress between the armour and structure. One candidate material for this interlayer is FeltMetal™ (Technetics Group, USA). This material consists of an amorphous matrix of fine copper wires which are sintered onto a thin copper foil, creating a sheet of approximately 1mm thickness. FeltMetal has for many years been successfully used to provide compliant sliding electrical contacts for the MAST TF coils and extensive material testing has therefore been undertaken to quantify thermal and mechanical properties. These tests, however, have not been performed under vacuum or DEMO-relevant conditions. A bespoke experimental test rig has therefore been designed and constructed with which to measure the interlayer thermal conductance as a function of temperature and pressure under vacuum conditions. The design of this apparatus and the results of experiments on FeltMetal as well as other candidate interlayers are presented here. In parallel, joint mockups using the candidate interlayers have been prepared and Thermal Break divertor target mockups have been manufactured, requiring the development of a dedicated joining process. These mockups will be subjected to high heat flux testing to further demonstrate the viability of the Thermal Break concept.

Id 865

Abstract Final Nr. P1.093

Enhancing the DEMO Divertor Target by Interlayer Engineering

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A robust divertor target plate solution for DEMO has to date remained elusive. Contemporary water-cooled concepts comprise a CuCrZr structural pipe (an acceptable material in the low neutron fluence divertor region), an interlayer, and tungsten armour monoblock. The interlayer, typically pure copper, facilitates joining of the parts and is often regarded as a compliance layer. In this paper we show by design optimisation and mock-up testing that an effectively designed interlayer can produce dramatic gains in power handling. CCFE has proposed a promising target concept termed “Thermal Break”. We reason that an effective interlayer should have a low bulk modulus and a relatively low thermal conductivity. These features allow the Thermal Break to reduce the thermal mismatch stress, redistribute heat flux around the pipe and place the materials at more favourable temperatures. The potential of the Thermal Break concept has been investigated using a design search and optimisation method based on Kriging response surfaces. The objective of the algorithm is to maximise the minimum reserve factor for a range of failure modes, over nine radial paths through the structure. This method has yielded a set of interlayer properties which, if realised, would give a DEMO-relevant design which passes all the ITER SDC-IC elastic design rules for an incident heat flux of 10MW/m2 with a structural safety margin of 1.4. The upper limit of power handling is 18MW/m2, corresponding to local plastic flow in the CuCrZr structure. Further, these optimisation results are feeding into a parallel programme of mock-up testing, in which Thermal Break mock-ups using “saddle” type armour have been manufactured. Through this coupled work programme, we are approaching a solution for a divertor target which meets the demanding requirements. This work was part-funded by the RCUK Energy Programme and by the European Union's Horizon 2020 research and innovation programme.

Id 959

Abstract Final Nr. P1.094

The WEST project: Qualification programme for the ITER divertor tungsten plasma facing component technology

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The ITER Council approved recently the proposal by the ITER Organization that ITER operation will start with a full tungsten divertor. However, the required actively cooled tungsten plasma facing technology has never been produced at large industrial scale, nor tested in the demanding environment of a tokamak under the steady state plasma heat fluxes expected in ITER (up to 20 MW/m2). In order to mitigate the risks for ITER divertor, Tore Supra, through the WEST program (W Environment for Steady-state Tokamak), will be progressively equipped with a full W-divertor representative of ITER divertor technology. The project supports the divertor strategy of ITER by providing the qualification of an industrial production of W-high heat flux components ahead of ITER series fabrication as well as by testing them for the first time in a tokamak environment, while taking advantage of the unique long pulse capabilities and the high power offered by the Tore Supra platform. This paper gives a brief description of the WEST full-W divertor, accounting for its relevance to the ITER one, and reports on the key aspects of the qualification program (i) during the manufacturing phase and (ii) during the operation phase. The WEST divertor manufacturing phase fills the gap between ITER prototype manufacturing and large series production (following CEA's former experience for its previous high heat flux plasma facing components and difficulties encountered in areas as diverse as non-destructive testing, optimization of manufacturing process, heterogeneous joints, tolerances, machining methods, etc.). WEST divertor manufacturing (~500 components representing ~14 % of ITER divertor surface) will provide sufficient statistics to assess the main manufacturing difficulties, optimize the acceptance procedures and ultimately the rejection rate. The WEST operation phase will offer a unique integrated qualification of actively cooled tungsten plasma facing components under relevant plasma conditions (e.g. steady-state heat fluxes in the range 10-20 MW/m2) by adapting the divertor configuration. Extended plasma exposure will provide access to ITER critical issues such as PFC lifetime (melting, cracking, etc.), impact of a large number of sub-threshold ELMs, effect of surface shaping and leading edges and study of surface evolution under high fluence plasma wall interactions well in advance of ITER divertor operation.

Id 431

Abstract Final Nr. P1.095

Design and Manufacturing of WEST Baffle

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The Tore Supra tokamak is being transformed in an x-point divertor fusion device in the frame of the WEST project, launched in support to the Iter tungsten divertor strategy. The WEST Baffle aims to evacuate particles by creating a groove between itself and the lower divertor. It also protects the lower divertor cooling pipes and the passive stabilization plate from heat fluxes. The Baffle is made of actively cooled Plasma Facing Components (PFC), a passive stabilization plate and a support beam. It has to be able to handle thermal and electromagnetical loads. Different types of lower divertor will be tested in the frame of the WEST project. Because of space constrains, a Baffle sector of 30° has to be removed each time a divertor sector of 30° is changed. As a consequence, its disassembly has to be fast and simple. Therefore hexagon socket head cap screws were chosen for the Baffle fixing. Their head is hidden from the plasma in counterbores, and they can be removed easily. Several 3D analyses are performed with ANSYS code. The optimal dimensions of PFC regarding electromagnetic loads and cooling are first investigated. The handling and fixing are then defined according to mechanical loads and facility of assembling/disassembling. After this pre-sizing phase, thermal analysis is performed to validate the cooling system and finally a plastic thermo-mechanical analysis is done to validate the fixing. The manufacturing of the PFC is then presented. The PFCs are made of CuCrZr, with a tungsten coating on the top surface. The cooling channels are drilled in the PFC, and electron beam welding is used to plug them. The cooling pipes are welded to PFC thanks to CuCrZr-INOX 316L connectors assembled thanks to explosion welding.

Id 540

Abstract Final Nr. P1.096

Heat flux depositions on the WEST divertor and first wall components

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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 made of 456 Plasma Facing Units (PFUs), to mitigate the risks for ITER. Concerning the others Plasma Facing Components (PFC), they will be also totally replaced by actively cooled components with a copper heat sink and a tungsten coating (upper divertor, baffle, VDE and ripple protections). However, since all the 456 PFUs will not be ready for the start of WEST in early 2016, inertial divertor PFUs will be used. They are made of graphite with a tungsten coating ~20 µm, similar to what has been developed for ADSEX upgrade. It will be replaced in a later step by a fully actively cooled divertor after 2 years. The key part of this paper shows simulations of the heat flux deposition pattern on the different PFC, for two extreme plasma scenarios: a close X-point configuration (distance X-point to target dX~1cm), and a far X-point configuration (dX~7cm). The heat flux is computed with the PFCFlux code for several power e-folding lengths in the Scrape-Off Layer (from 2 to 10mm in the mid-plane). Some complexity is also added by taking into account the ripple of the magnetic field due to the discrete nature of the superconducting toroidal field coils, which is rather large on Tore Supra (few % in the equatorial plane) and plays an important role on the heat flux deposition pattern. Others sources of heat fluxes, such as ripple losses, are also evaluated. The second part of this paper describes the general concept of each kind of PFC (material choices, manufacturing process…), and their limitations in terms of input power. The paper concludes on the adequacy of the foreseen heat loads, the PFC concepts and the experimental program.

Id 600

Abstract Final Nr. P1.097

Leak tightness tests on actively cooled plasma facing components: lessons learned from Tore Supra experience and perspectives for the new fusion machines

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In order to minimize the risks of defects on actively cooled Plasma Facing Components they have to be manufactured with QA requirements and tested before their installation in the vacuum vessel. Their leak tightness qualification is part of thes tests. The proposed paper will report on the feedback gained at Tore Supraon this topics and point out the points, issues and procedure improvements which can be useful for the ITER components.

Id 634

Abstract Final Nr. P1.098

Plasma Facing Components integration studies for the WEST divertor

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In the context of the Tokamak Tore-Supra evolution, the CEA aims at transforming it into a test bench for ITER Actively Cooled Tungsten (ACW) plasma facing components (PFC). This project named WEST (Tungsten Environment in Steady state Tokamak) is especially focused on the divertor target. The modification of the machine, by adding two axisymmetric divertors will make feasible an H-mode with an X-point close to the lower divertor. This environment will allow exposing the divertor ACW components up to 20MW/m² heat flux during long pulse. These specifications are well suited to test the ITER-like ACW target elements, respecting the ITER design. One challenge in such machine evolution is to integrate components in an existing vacuum vessel in order to obtain the best achievable performance. The divertors coils are designed regarding the magnetic specifications, the plasma facing components are placed according to the plasma shape, and then the interfaces have to be managed regarding the remaining space. Moreover in this layer, many important smaller components have to be integrated as cooling pipes, magnetic diagnostics, gas injection, Langmuir probes, thermocouples, etc. This paper deals with the design integration of ITER ACW target elements into the WEST environment considering magnetic, electric, thermal and mechanical loads. The feasibility of installation and maintenance has to be strongly considered as these PFC could be replaced several times. The ports size allows entering a 30° sector of pre-installed tungsten targets which will be plugged as quickly and easily as possible. The main feature of steady state operation is the active cooling, which lead to have many embedded cooling channels and bulky pipes on the PFC module. It means to have many connections and sealings between vacuum and water channels. The 30° sector design is now finalised regarding the ITER ACW elements specifications. No major modifications are expected.

Id 648

Abstract Final Nr. P1.099

Tungsten coating developments on large size and complex geometries CuCrZr elements for the WEST project

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The WEST (W Environment in Steady-state Tokamak) project consists in implementing a divertor configuration and installing an ITER like actively cooled tungsten divertor in the Tore Supra tokamak, taking full benefit of its unique long pulse capability. A significant transformation of all the Plasma Facing Components (PFC) of Tore Supra (carbon to tungsten) is required. In WEST, the lower divertor will use the ITER technology of W monoblocks able to sustain heat fluxes in the range 10 MW/m2 in steady state. Other in-vessel components (the upper divertor, the baffle and the ripple protections) will be based on copper heat sinks covered with W coatings and are foreseen to handle heat fluxes up to 5 MW/m2 in steady state. In order to propose high quality W coatings on large size CuCrZr elements (up to 600*120mm) with complex geometries (rounded edges), several deposition technics have been evaluated (CVD : Chemical Vapor Deposition; PVD : Physical Vapor Deposition; HPPS : High Pressure Plasma Spray). One parameter to be optimized is the coating thickness. While a significant W coating thickness would be favorable to increase coating lifetime, a lower thickness would enhance the mechanical bonding cohesion as it was demonstrated for the JET and ASDEX Upgrade tungsten coatings development. In order to find a tradeoff between coating lifetime and coating integrity during WEST operation, several W thicknesses (from 15µm to 2mm) have been compared in terms of physical analysis (thickness uniformity on complex geometries, density, composition, roughness) and tested under cyclic high heat fluxes. The correlation between the deposition techniques, the physical characterization and the coating behavior under cyclic heat fluxes was investigated and will be reported in this paper.

Id 539

Abstract Final Nr. P1.100

Status of the WEST actively cooled upper divertor

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The WEST (W -for tungsten- Environment in Steady-state Tokamak) project is based on an upgrade of Tore Supra, transforming it in an X-point divertor device, while taking advantage of its long discharge capability. As in any tokamak, plasma-facing components (PFCs) must provide adequate protection of in-vessel structures. This function takes on particular significance in WEST tokamak, as in ITER, which will combine long pulse (>1000s), high power operation (up to 15 MW for WEST) with severe restrictions permitted core impurity concentrations and which, in addition, will produce transient energy loads. The new configuration will allow for H-mode access, providing relevant plasma conditions for PFC technology validation in a totally metallic environment. This paper presents the upper divertor of WEST project (design constraints, thermo-mechanical performances). Upper divertor must be able to operate with double null or upper single null equilibria. This component, with a total surface of 8 m2, has to exhaust 4 MW of conducted power in steady state (reference equilibriums with strike points on the target). As the heat exhaust is important, this component is an actively cooled target with a CuCrZr heat sink. In order to be consistent with PFCs covered with metallic armour materials, upper divertor is covered with a W coating. Cooling channels and hydraulic conditions are chosen as a compromise between hydraulic/mechanical/fabrication constraints to rich the adequate thermal performance. The component and its mechanical attachments are designed to resist to mechanical loads which are mainly due to possible Vertical Displacement Event (dBn/dt=-90Ts-1, dBs/dt=-70Ts-1, during t=10ms). The proposed concept enables to exhaust locally up to 10 MW/m2 in steady state.

Id 925

Abstract Final Nr. P1.101

Optimization of the First Wall for the DEMO Water Cooled Lithium Lead Blanket

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The maximum heat load capacity of a DEMO First Wall (FW) of reasonable cost may impact the decision of the implementation of limiters in DEMO. An estimate of the engineering limit of the FW heat load capacity is an essential input for this decision. This paper describes the work performed to optimize the FW of the Water Cooled Lithium-Lead (WCLL) blanket concept for DEMO fusion reactor in order to increase its maximum heat load capacity. The optimisation is based on the use of water at typical Pressurised Water Reactors conditions as coolant. The present WCLL FW with a waved plasma-faced surface and with circular channels was studied and the heat load limit has been predicted with FEM analysis equal to 1.0 MW.m-2 with respect to the Eurofer temperature limit. An optimization study was then carried out for a flat FW design considering thermal and mechanical constraints assuming inlet and outlet temperatures equal to 285°C/325°C respectively and based on geometric design parameters such as channel pitch, diameter of pipes and thicknesses. It became clear through the optimization that the advantages of a waved FW are diminished. Given the manufacturing issues of that concept, the waved FW was therefore not pursued further. Even if the optimization study shows that the maximum heat load could in principle be as high as 2.53 MW.m-2, it is reduced to 1.57 MW.m-2 when additional constraints are introduced in order not to affect corrosion, manufacturability and Tritium Breeding Ratio in normal condition such as a coolant velocity ≤8m/s, pipe diameter ≥5 mm and a total FW thickness ≤22mm. However it is important to note that the FW channels currently fulfil additional functions and are therefore not optimized “at all cost” regarding heat load capacity and the paper points out some recommendations against missing assumptions.

Id 135

Abstract Final Nr. P1.102

In-pile testing of ITER first wall mock-ups at relevant thermal loading conditions in LVR-15 research reactor

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Under EFDA contract TW3-TVB-INPILE, Centrum výzkumu Řež has carried out a number of testing and R&D activities in support of the development of the ITER plasma facing components (FW panels). This paper describes the neutron irradiation campaign. In order to assess the material properties under ITER relevant operational conditions it is of great importance to investigate the influence of irradiation, especially on the Be/CuCrZr Hot Isostatic Pressing (HIP) joints. Two Primary First Wall (PFW) mock-ups were irradiated to 0.46 dpa on Beryllium (0.6 dpa CuCrZr/SS heat sink), with parallel thermal cyclic fatigue testing at 0.5 MW/m2 for 17,040 cycles. Each cycle was 6,5 minutes long and the whole test covered of 109 full-power days of irradiation. An electric heater made of high temperature graphite was used to provide the cyclic thermal load onto mock-up surface. The development of the TW3 experimental rig is described, as well as the technique used to perform in-pile thermal fatigue testing of force-circulation cooled PFW mock-ups inside the core of research reactor LVR-15.

Id 250

Abstract Final Nr. P1.103

Computational Thermal Fluid Dynamic Analysis of Hypervapotron Heat Sink For High Heat Flux Devices Application

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In fusion devices, plasma is the environment in which light elements fuse producing energy. About 20% of this power reaches, the surface of Plasma facing components (especially the Divertor targets, Firstwall), where the heat flux local value can be several MW/m2. In order to handle such heat fluxes several coolants are proposed such as water, Helium and Liquid metals along with different heatsink devices, such as Swirl tubes, Hypervapotrons, Jet cooling, Pin-fins, roughness [1, 2]. Among these, Hypervapotron concept, operating in highly subcooled boiling regime with water as a coolant is considered as one of the potential candidates [2]. In this paper a Computational Fluid Dynamic (CFD) approach is used to analyze the boiling flow inside Hypervapotron channel using two different boiling models: the Rohsenow boiling model [3], with the capability to model both nucleate and film boiling regimes, which was previously tested on flat-channel geometry [4] and Transition boiling model [3], with the capability to model nucleate and transition boiling regimes, which is more general than Rohsenow model. These models are available in the commercial CFD code STARCCM+ [3], and uses Volume of Fluid approach for the multiphase flow analysis. They are benchmarked using experimental data obtained from experiments conducted on Hypervapotron at Joint European Torus, UK. The simulated results are then compared against each other and also with simulated data by J.Milnes [5] to test the quantitative, qualitative features of boiling models in modelling nucleate as well as hard boiling regimes [5]. 1. C.B. Baxi et.al, Fusion Engineering and Design 51–52 (2000) 319–324. 2. R. Mitteau et.al, Fusion Engineering and Design 88 (2013) 568– 570. 3. STAR-CCM+ version 7.02 user's guide, CD-apdapco.inc., (2012). 4. Phani Domalapally et.al, Fusion Engineering and Design 87 (2012) 556– 560. 5. J. Milnes et.al, Fusion Engineering and Design 87(2012) 1647–1661.

Id 74

Abstract Final Nr. P1.104

The high-heat-flux test facilities in the Efremov institute

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There are two electron beam HHF testing facilities in the Efremov Institute. The first is the TSEFEY-M facility was put into operation in 1994. Main purposes of this facility is thermal fatigue testing of different mock-ups with different facing materials (CFC, tungsten, beryllium and etc.) with different cooling type (water or gas). For the period from 2008 to 2014, various mock-ups with different armor (tungsten, CFC and beryllium) were subjected to high heat flux tests on the TSEFEY-M test facility. The surface heat load density amounted to 20 MW/m2. At present time TSEFEY-M facility is able to test wide range of water cooled beryllium-armored mock-ups and small helium cooled mock-ups. Incident power up to 200kW is available, water loop can provide inlet pressure more than 4MPa, inlet temperature up to 170C and mass flow rate up to 4 kg/s. The second one is the ITER Divertor Test Facility (IDTF). This facility was created for the main activities on the procurement arrangement 1.7.P2D.RF. (high heat flux tests of the plasma facing units of the ITER divertor). In the Efremov Institute the IDTF was created. At the end of 2012 on the IDTF the first significant task of the PA 1.7.P2D.RF. – the high heat flux testing of the test assembly of the outer vertical target full-scale prototype – was completed. IDTF facility is able for the high heat flux tests of the PFUs of the outer vertical targets, the inner vertical targets and the domes. Incident power up to 750kW is available, water loop can provide inlet pressure more than 4MPa, inlet temperature up to 140C and mass flow rate up to 10 kg/s. The diagnostic complex of both facilities includes IR-cameras, pyrometers, thermocouples and appropriate water loop diagnostics.

Id 876

Abstract Final Nr. P1.105

Upgrade of EAST In-vessel Components

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The EAST plasma heating power is being increased to 20MW and further increased to 30MW in few years. Heat load to divertor will be increased to 10MW/m2 or more for steady state, type-I ELM has been observed in last plasma campaign operation. To design and develop new divertor to handle high heat load, and to install resonant magnetic perturbation (RMP) coils to mitigate ELM are necessary. EAST upper divertor is updated to tungsten copper divertor with tungsten PFM and lower divertor is still keep in carbon PFM. ITER divertor tungsten monoblock technology was applied for EAST new divertor. New divertor was designed as module structure with tungsten copper target attach to stainless steel cassette body. There are 80 module for total, and all modules were supported by rails installed in vacuum vessel. Monoblock technology was applied for strike point target and flat plate tungsten copper bonding technology was applied for baffle and dome. HIP technology was employed for materials bonding, i. e. tungsten bonded to pure copper and pure copper bonded to CuCrZr. Electron beam welding was applied for monoblock units connect to flat plate. Testing has been made demonstrated monoblock mock up can sustain 10MW/m2 for more than 1000 cycles, and flat plate mock up can sustain 5MW/m2 for more than 1000 cycles. RMP coils are located at outer part of vacuum vessel up-down symmetric. Eight coils upper and eight coils down. Conductor for coils is water cooling oxygen free copper with magnesium oxide insulation and stainless steel jackting. Besides, VS coils with same RMP coils conductor technology are relocated for higher instability control efficiency, and additional in-vessel cryo-pump installed for upper divertor pumping. Updated EAST in-vessel components are expected to operate around May of 2014. The paper will present new components design, development, installation and preliminary operated results.

Id 174

Abstract Final Nr. P1.106

Concept design of the CFETR divertor remote handling system

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The Chinese Fusion Engineering Testing Reactor (CFETR) is a superconducting tokamak fusion device envisioned to provide 200 MW fusion power. Because of erosion with hazardous and/or activated dust (beryllium, carbon, tungsten) and tritium, the use of remote handling techniques will be eventually required. In particular, the replacement of components such as divertor modules will require the use of remote cutting, welding and inspection of cooling pipes. CFETR consists of 60 divertor cassettes located in the bottom region of the vacuum vessel. The cassettes will be withdrawn from the vessel through four dedicated ports and transported to a hot cell for refurbishment and replacement during the reactor's operational lifetime. The divertor system remote replacement is one of the key maintenance operations for the CFETR due to the requirements of duty factor (0.3~0.5), machine safety, high plant availability and reliability. The critical issues such as the operational conditions (temperature, atmosphere, radiation, contamination), functions, mechanical design requirements of the divertor Remote Handling Maintenance System (RHMS) are described. The concept design of the remote handling equipment, the operating procedures and the maintenance time estimations for high plant availability are also discussed in this paper.

Id 220

Abstract Final Nr. P1.107

Thermal Testing of the ITER Diagnostic Cable Loom

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In ITER, electrical cables shall be used to transmit signals of the numerous mainly magnetic and bolometer diagnostic sensors from the vacuum vessel inner wall to the air side of the tokamak. These cables are collected in cable looms, housing 25-41 cables. The looms are located between the VV inner wall and blanket modules and subject to high radiation levels, high temperatures and high magnetic field. It is essential that the temperature does not vary more than 10K along a specific cable length, so that the transferred signals remain accurate. The nuclear heat in the cables is conducted into the water cooled vacuum vessel wall. The heat distribution in the looms is highly dependent on the resulting contact surface between the vacuum vessel wall and the loom bottom, which is an aluminium plate. The main problem is that the contact between the VV wall and the aluminum plate cannot be flush, as the VV inner surface is a complicated 3D surface of a torus, and the aluminum plates are flat bottomed. The contact surface between them shall be depending on compression force and exact location of the loom section. Because of the many variables, the thermal conductance coefficient cannot be determined reliably by calculation, only with testing. This paper presents a test in which ITER conditions are simulated. In a vacuum chamber a small section of the VV wall with the most critical curvature is used, the bottom of which is water cooled and the nuclear heat is simulated by ohmic heated MI cables. The short loom section (design based on IO concept) is attached on the surface and thermal conductance coefficient measured with temperature sensors. This test allowed us not just to determine if a loom design satisfies the strict condition, but also to compare and optimize several designs.

Id 462

Abstract Final Nr. P1.108

Progress on DEMO Blanket Attachment Concept with Keys and Pins

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The blanket attachment has to cope with gravity, thermal and electromagnetic loads, also the blanket has to be installed and serviced by remote handling. Pre-stressed components suffer from stress relaxation in irradiated environment such as DEMO. Therefore attaching the blanket modules to the back plate and the back plate to the vacuum vessel via bolted connections would lose large amount of their pre-load. To circumvent this problem pre-stressed component should be either avoided or shielded. Alternatively the blanket segments could be attached to the vacuum vessel by a set of keys and pins. This strategy has been proposed for the DEMO multi module segments. The blanket segments are held by two tapered keys each, designed to allow thermal expansions while providing contact with the vacuum vessel and to resist the poloidal and radial moments the latter being dominant at 9.1 MNm inboard and 15 MNm outboard. On the top of the blanket segment there is a pin which provides vertical support. At the bottom another vertical support has to lock the blankets in position after installation and manage the pre-load on the blankets. The pre-load is required to deal with the electromagnetic loads during disruption. This could be a set of springs, which require shielding as they are pre-loaded. These are sized to cope with the force (3MN inboard, 1.4 MN outboard) due to halo currents and the toroidal moment; the disruption loads can reverse. Calculations show that the flexibility of the blanket segment itself plays a huge role in defining the required support system. The blanket acts as a pre-loaded spring and it has to be part of the attachment design as well. This work was funded by the RCUK Energy Programme under grant EP/I501045 and by the European Communities under the contract of Association between EURATOM/CCFE.

Id 608

Abstract Final Nr. P1.109

The 'ductility exhaustion' method for static strength assessment of fusion structures

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The traditional approach to check the static strength of fusion structures involves the computation of elastic stresses followed by an assessment against local and global failure using stress categories. At the heart of this route is the concept of “linearised stress lines” where failure is associated with critical through-thickness ligaments. For fusion structures typically analysed with finite elements (FE), problems can arise since: • These structures are often not shell-like and lack well defined ligaments. • The assessment stage can be lengthy and overly conservative. • Assigning stress categories can be difficult and sometimes arbitrary. Now however, highly realistic plastic simulations are possible where local failure can be simply defined as exhaustion of ductility when peak plastic strains exceed available plastic ductility which in turn depends only on the tri-axial stress factor (TF or ratio of mean tension to Von Mises stress). Some design codes (e.g. ASME VIII 2013 edition) now include this “ductility exhaustion” approach which also gives a useful insight into the failure modes such as tension-necking or compressive shear distortion depending on the TF distribution. The method has been benchmarked for both a tension case (“dog-bone” specimen) and a compression case (short cylinder upset) using both the Abaqus and ANSYS FE codes. It has then been applied to one of the JET beryllium tiles assemblies where the results are less conservative, at 600 C, by a factor of about 2 compared with the traditional method. Furthermore, the method shows that the minimum JET operating temperature, currently 200 C. could be no lower than about 120 C, if failure due to reduced ductility of beryllium is to be avoided. This work was part-funded by the RCUK Energy Programme and by the European Union's Horizon 2020 programme.

Id 846

Abstract Final Nr. P1.110

Remote Handling assessment of attachment concepts for DEMO blanket segments

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The replacement strategy of the massive Multi-Module Blanket Segments (MMS) is a key driver in the design of several DEMO systems. These include the blankets themselves, the VV and its ports and the Remote Maintenance System (RMS). Under EFDA WP13 DAS02 activities two different attachment concepts were produced for fixing the MMS to the Vacuum Vessel (VV). The Remote Maintenance DAS07 task 5 carried on the development of feasible RH operations and equipment considering both support concepts at different stages of their design. Common challenges have been identified, such as the need for applying a preload to the MMS manifold, the effects of the decay heat and several uncertainties related to permanent deformations when removing the blanket segments after service. The WP12 kinematics of the MMS in-vessel transportation was adapted to the requirements of each of the supports. A shared outcome recommendation for avoiding clashes is the modification of the central outboard MMS segmentation, which would also allow future changes in the basic geometric parameters of the blanket segments if required. An RMS compatible geometry for blanket segments—including overlapping features between them—has been also proposed in case the neutronic studies indicate that it is needed for protecting the VV, ensuring its compatibility with both support designs. The RM equipment envisaged for handling the MMS, its attachments and its earth connections includes three different systems. An In-Vessel Mover at the divertor level handles the lower support and earth bonding, and stabilizes the MMS during transportation. A Shield Plug crane with a 6 DoF manipulator operates the upper attachment and earth straps. And a Vertical Maintenance Crane is responsible for the in-vessel MMS transportation and can handle the removable upper support pins.

Id 879

Abstract Final Nr. P1.111

Application of virtual reality tools for assembly of WEST components: Comparison between simulations and physical mockups

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The WEST project (Tungsten (W) Environment in Steady state Tokamak) is an up-grade of the existing fusion machine Tore Supra. The goal is to equip the tokamak with a fully cooled tungsten divertor and to transform the machine in a test platform open to all ITER partners. The main assembly challenge of this project consists to the implementation of two divertors with an accuracy of 1 millimeter. Indeed, each divertor is about 4 meters diameter and a heavy weight of 10 tons, introduces piece by piece in the original vessel trough tight ports then assembled inside. To ensure a perfect fitting between this new components and a very constrained environment, it is necessary to use the latest CAD technologies available. Beyond conventional CAD tools, the virtual reality (VR) room of the institute provides several useful tools. Thanks to the 185” stereoscopic 3D screen and a force feedback arm linked to clash detection software developed by the CEA LIST, a new way to carry out design and assembly studies was performed. In order to improve VR results, metrology data (3D scan) enhance simulations. Therefore, it becomes possible to be aware of the real size of a component and future difficulties to assembly it. At last, performance of such simulations are evaluated and compared to physical mockup in order to bring enhancement to the VR tools, before to be compared to the real operations on Tore Supra. The aim is to build a design tool that help the designer since early stage of the design of complex systems, taking into consideration integration, assembly and maintenance aspects while reducing costs and schedule of a project.

Id 497

Abstract Final Nr. P1.112

Major upgrade of the Articulated Inspection Arm control system to fullfill daily operation requirements

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An Articulated Inspection Arm (AIA) has been developed by CEA for visual inspection between pulses inside the Tore Supra Tokamak vacuum vessel without breaking temperature and vacuum conditions. The height meters length robot is composed of a shuttle and six articulated segments with a video camera at its end. A demonstration prototype has been achieved in 2008 at Tore Supra. A project to upgrade the AIA into a fully operational robot has been set between IRFM and ASIPP in an Associated Laboratory. It will be in operation first in the EAST machine and after wards in Tore Supra in its WEST (W/Tungsten Environment in Steady-state Tokamak) configuration where it is of paramount importance to survey possible degradation of W component surface. The control system of the robot has been extensively upgraded. The effort has been focused on three areas: 1) improvement of the arm position accuracy, 2) increase of the operational robustness, 3) use of a powerful graphical user interface including simulation of trajectories and robot deployment capabilities in a 3D viewer environment. Several specific electronic modules have been developed. The control architecture has been done through a modular approach involving embedded and remote electronic controllers, and the use of CAN bus (Controller Area Network) and Ethernet communications. A special effort has been made on the implementation of the latest generation of position sensors. This electronic is coupled with new supervisor software, built on Actin, a commercial off-the-shelf package, enriched with specific functionalities. The supervision simulates the deployment preparation in a 3D model of the Tokamak, using reusable scenarios. Accurate and safe deployments are ensured by a collision avoidance monitoring and by a flexible model automatic correction based. The aim of this paper is to detail the architecture of the AIA control system and to present the obtained results.

Id 530

Abstract Final Nr. P1.113

The portable metrology on tore supra for the design & assembly optimization of the west components

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In the framework of the WEST project, Tore Supra tokamak will be equipped with a lower and upper divertor allowing to test the ITER divertor target elements in realistic flux conditions during long pulses. The major part of the Plasma Facing Components (PFCs) will be replaced. To validate the mechanical interfaces and maximize the plasma volume, the WEST inner components must be set up in the vacuum vessel with a great accuracy. The geometry of the vessel and of the ports is presently known with an accuracy of 5 to 10 mm, which is clearly not enough for the WEST project requirements. In 2013, dedicated metrology campaigns have been defined and carried-out to compare the 3D CAD models of vacuum vessel with the As-built geometry using an accuracy of 0,1 mm and transfer temporarily the inner vessel walls referential during the dis/assembly phases. New metrology techniques (virtual metrology, portable metrology) and equipments (laser tracker, T-Scan,…) have been used which allowed to reach these goals. Other metrology campaigns are planned to position all of the WEST components (axisymmetric divertor structure, protection panels, plasma facing component,…) inside the vacuum vessel during the reconstruction phase. The CEA/IRFM works in close collaboration and with support of SETIS GROUPE-DEGAUD Company which is specialized in multipurpose metrology. This collaboration allows us to get assistance for the integration of the components and define the assembly procedures for the PFCs integration. The paper will present the main results got during the already achieved campaigns of metrology in the disassembly phase in 2013, and detail the methods which have been used. The next metrology campaigns to assemble the components of WEST will be addressed.

Id 688

Abstract Final Nr. P1.114

Manufacturing monitoring and mock-ups validation 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, the implementation of an axisymmetric divertor structure in the tokamak Tore-Supra has been studied. With this major upgrade, so called WEST (Tungsten Environment in Steady state Tokamak), Tore-Supra will be able to address the issues of long plasma discharges using a tungsten divertor based on monoblock targets. The divertor structure and coils assembly is made up of two stainless steel casings containing a copper winding pack cooled by a pressurized hot water circuit (up to 180°C, 4MPa) in which a current of 11.7kA is circulating. The conductor is electrically insulated by Kapton layer (20 mm wide, 3 layers of 67µ, 50% overlapping) and locked inside the casing thanks to epoxy shims in order to be mechanically protected. The divertor is designed to perform steady state plasma operation (up to 1000s), it must sustain harsh environmental conditions in terms of ultra hight vacuum conditions, electromagnetical loads and electrical isolation (13 kV ground voltage) under high temperature (180°C). Therefore feasibility study of such a complex structure has been performed. It implies activities on a scale one dummy coil, such as installation, assembly and representative tests (electric, thermal and hydraulic). In parallel a scale one mechanical connection between two casings has been produced. The manufacturing of the divertor structure, which is a large assembly of 4 meter diameter representing a total weight of around 20 tonnes, started in the second half of 2013 and is to be delivered by the end of 2014. The paper will illustrate the technical developments and test performed during 2013 and beginning of 2014 in order to fully validate the design before the industrial phase. The manufacturing methods proposed by the contractor in order to fulfil the technical requirements will be also addressed.

Id 784

Abstract Final Nr. P1.115

Design of WEST divertor coils

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Tore Supra is upgraded to become a test bed aimed to fully validate actively cooled tungsten plasma facing components (PFC). During this major upgrade, so called WEST (Tungsten Environment in Steady state Tokamak), two divertor coils will be inserted in the vacuum vessel to create an X-point magnetic configuration able to provide ITER relevant conditions for the divertor target. The paper will illustrate the analysis performed to obtain the current design: from the material choice to the definition of the limit parameter of the coils. Conductors are made of copper-silver alloy cooled by pressurized water (up to 200°C, 4MPa). In steady state operation (up to 1000s), a current of 11,7kA flows in the spires which generates thermal stress due to the heating power dissipated by Joule effect. The fluid velocity of 5m/s is sufficient to keep a thermal gradient of 50°C. In order to reduce the thermal gradient between the windings and the casing (at 70°C), the inlet temperature of the fluid is optimized to 50°C. Some thermomechanical analyses are performed with the ANSYS code to validate the sizing of the different parts of the turns, like water renewal inlets/outlets and spire changes. The electrical inlets/outlets are stressed by electromagnetical loads. The toroidal magnetic field (up to 5T) interact with the conductors. It's necessary to foresee some shims in the casings according to mechanical calculations. This year, some tests will be performed on mock-ups including the main design parameters in order to fully validate the design before the industrial phase.

Id 813

Abstract Final Nr. P1.117

Progress on the design of a brazing connector for DEMO in-vessel components

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The biggest unavailability sources in a fusion power plant will be caused by the replacement of large in-vessel components (breeding blanket modules and divertor cassettes), as well as by components failure and short duration maintenance tasks and inspection processes. Present maintenance procedures are based on current technology solutions for equipment adapted from other uses. The development and validation of remotely operable leak-free pipe connectors for helium, water and liquid metal heat transfer fluids is a key issue for faster and more reliable replacement of in-vessel components. This work further develops the design of a connector by brazing for DEMO blanket pipes proposed within the EFDA Task WP12-DAS06-T06. Two alternative clamping concepts to stiffen the connector are proposed here, as well as a RH device to install/uninstall them. Furthermore, the mechanical behaviour of the joint between the Ni-200 parts through the BAu-4 filler metal is characterized during a cooling scenario after brazing. Ni-Pd-Cr alloys are proposed as an alternative to BAu-4 as filler metal, considering their mechanical properties are comparable to the BAu-4 ones. In addition, different previous works show brazed joints using BAu-4 and BNi-5 keep the mechanical properties of the original filler metal, which may be superior to the base metal ones. This suggests the possibility of removing the mechanical clamp, which would improve the integrability of the brazed connector. On the other hand, the capillary flow of the BAu-4 filler metal during the brazing process is modelled using a CFD approach, in order to establish a comparison with analytical models results. Finally, a preliminary assessment on the issue of tritium permeation is made.

Id 565

Abstract Final Nr. P1.118

Innovative design for FAST divertor compatible with remote handling, electromagnetic and mechanical analyses

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Fusion Advanced Studies Torus (FAST) is conceived as a satellite tokamak acting as test bed for the study and the development of innovative technologies oriented to ITER and DEMO programs. Divertor is a crucial component in Tokamaks, aiming to exhaust the heat power and particles fluxes coming from the plasma during discharges. Divertor shall be replaceable as well during machine lifecycle. This is made by means of a proper Remotely controlled Handling system (RH). FAST Divertor RH system is mainly composed of two subassemblies, the Cassette Multifunctional-Mover and the Second Cassette End-Effector, the latter equipped with a further arm whose function consists in performing the critical operations of locking and unlocking the cassette in its relative support system. Divertor RH system final layout has been chosen between different concept solutions proposed and analysed within the principles of Theory of Inventive Problem Solving (TRIZ). The design of the RH is aided by kinematic simulations performed using Digital Mock-Up capabilities of Catia software. FAST divertor structure is compact and aims to achieve required thermo-mechanical capabilities and to be RH compatible. Considerable electromagnetic (EM) analysis efforts and top-down CAD approach enabled the design of a final and consistent concept, starting from a very first dimensioning for EM loads. In the final version here presented, the divertor cassette supports a set of tungsten (W) actively cooled tiles which compose the inner and outer vertical targets, facing the plasma and exhausting the main part of heat flux. W-tiles are assembled together considering a minimum gap tolerance (0.1÷0.5mm) to be mandatorily respected. Cooling channels have been re-dimensioned to optimize the geometry and the layout of coolant volume inside the cassette has been modified as well to enhance the general efficiency. Consistency of the design has been then confirmed by further Finite Element EM and mechanical analyses.

Id 886

Abstract Final Nr. P1.119

Artificial Landmark based Localization of Tokamak Transfer Cask System

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The Transfer Cask System (TCS) is a critical element of the ITER Remote Handling (RH) Maintenance System devoted to transportation of components between Tokamak Building (TB) and Hot Cell Building (HCB). It will be autonomously guided following predefined trajectories. Therefore, the localization of TCS in operation must be continuously known in real time to provide the feedback for the control system and also for the human supervision. A localization method based on electromagnetic guidance proposed before, which needs to install a guidance wire in the ground, has disadvantages in TB/HCB situation. There are three floors in TB, including upper port, equatorial port, and lower port, linked by a lift. To install the guidance wire between the floors is unprocurable. And the Tokamak building is made of steel reinforced concrete which has steel in it, however, metal around the guidance antenna will lead to signal attenuation and changes in the distance output so that the precision of localization cannot be guaranteed. Another method, SLAM, can be considered in such situation, but its computation complexity and lower precision are unsatisfactory. Besides, its result can be easily disturbed by moving objects. This paper proposes an artificial landmark based localization method that uses the scan data returned from the laser radar installed on the cask. Landmarks are deployed in TB previously, the laser radar can figure them out from the scan data according to reflectivity and build a landmark map. As the cask moves in the building, the system can calculate its global position and orientation by matching the scanned landmarks with the landmark map. The experiment results of this system showed that the localization error could be limited in 4mm in xy-direction and 1.5° in heading angle, and the proposed system with high robustness is flexible to various scenarios.

Id 384

Abstract Final Nr. P1.120

Validation of the remote handling refurbishment process for the European IFMIF Target Assembly concept design

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The remote handling maintenance of components of IFMIF facilities is one of the most challenging activities to be performed to guarantee the required high level of IFMIF plant availability. Among these components the maintenance of target assembly system appears to be critical because it is located in the most severe region of neutron irradiation. The present European target assembly design is based on the so called replaceable backplate bayonet concept. It was developed with the objective to reduce the waste material and to simplify the procedures for the target and backplate replacement and thus reducing the intervention time for their substitution. The remote handling maintenance activity of the target assembly comprises a number of in situ refurbishment tasks, like the: removal of the backplate, cleaning of surfaces from lithium solid deposition, inspection of the target body, installation of a new backplate and testing of the assembled system. However there is also the possibility to replace the entire target assembly and to perform these refurbishment tasks offline in a dedicated hot cell. To accomplish all the refurbishment operations for the target assembly within the expected time for maintenance, the annual preventive maintenance period for IFMIF has been fixed in 20 days, experimental activities aimed at validating the implemented maintenance procedures for this component was set up at the research centre at ENEA Brasimone (Italy). Both scenarios, the in situ and the offline refurbishment processes, were tested and the feasibility of each maintenance operation was proved. In the paper a description of the validation activities carried out together with the main outcomes obtained are given.

Id 1010

Abstract Final Nr. P1.121

Nuclear Analysis of the ITER Cryopump Ports

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The cryo-pump system is a fundamental component to ensure optimum plasma performance during ITER operations: its main function is to reduce the pressure inside the vessel, moreover it's designed to extract the Helium ashes generated by the deuterium-tritium fusion reactions, that will be processed in the tritium plant and fed back to the plasma. The ITER machine will be equipped with 6 torus cryo-pumps (TCP) that are positioned in their housings (TCPHs), integrated into the cryostat walls at B1 level in the port cells. The aim of the study presented in this paper, is to perform a complete assessment of nuclear responses in the Cryopump Ports #4 and #12 by means of the MCNP-5 Monte Carlo code in a full 3-D geometry. The results of the neutronic analysis provide guidelines for the design of the embedded components, in order to ensure their structural integrity, proper operations and to assess the shielding capabilities. The geometrical features of the Lower Ports have been singularly incorporated in the last MCNP ITER reference model, B-lite v3, including the Shielding, TCP structure, the TCPH and all the main surrounding components that are present in the area. Furthermore, the obtained models have been extended beyond the Bioshield to the port door, integrating details of the port cells and building. Radiation transport calculations have been performed in order to determine the radiation field inside the Lower Ports, up the port cells: 3-D neutrons and gamma maps have been provided in order to evaluate the shielding effectiveness of the TCPHs. Nuclear heating induced by neutron and photons have been estimated on the TCP and TCPH to assess the nuclear loads during plasma operations. The shutdown dose rate in the maintenance area of the Lower Ports has been assessed with the Advanced D1S method to verify the design limits.

Id 249

Abstract Final Nr. P1.122

Nuclear Analysis of ITER Test Blanket Module Port Plug

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Three of the ITER equatorial ports are dedicated to the Test Blanket Modules (TBMs) to test and validate design concepts. The TBMs Port Plug (PP) consists of a frame and two TBM/shield sets. The frame shall withstand heat loads and provide adequate neutron shielding for the vacuum vessel and magnets. The function of the dummy TBM is to replace the actual TBM-set, if the latter is not ready for its timely installation or in case of failure. The design of the frame and dummy TBM is under ITER responsibility. The assessment of the nuclear responses in the TBM PP components is important to guarantee the structural integrity and its proper operation. Furthermore, the TBM PP structure must contribute to the nuclear shielding of the port interspace where human access is needed for maintenance. Nuclear analyses have been performed with MCNP to support the design of the frame and dummy TBM. Neutron fluxes, nuclear heating, He production and neutron damage have been calculated in all the TBM PP components. Global shutdown dose rate calculations have also been performed with Advanced D1S method for different configurations of the TBM PP system. One critical issue of the current design is the dose rate in the maintenance zone that may exceed the design target of 100 µSv/h at 12 days after shutdown by the end of ITER life. The performed analyses show that the adopted double dog-leg configuration of the frame decreases the neutron streaming through the gaps between the components, but cannot guarantee the fulfillment of the design target mainly because of blanket configuration and crosstalk effect with upper and lower ports. This paper presents the results of these analyses and discusses the ongoing design improvements aimed to further reduce the shutdown dose rate in the maintenance zones.

Id 380

Abstract Final Nr. P1.123

IVVS probe mechanical concept design

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ENEA has being deeply involved in the design, development and testing of a laser based In Vessel Viewing System (IVVS) required for the inspection of ITER plasma-facing components. The IVVS probe shall be deployed into the vacuum vessel from its storage position, between plasma operations, providing high resolution images and metrology measurements to detect damages and to evaluate possible erosion. It must be therefore compatible with ITER environmental conditions such as high vacuum, gamma rays, neutron fluence, temperature of 120°C and magnetic field of 8 T. ENEA already designed and manufactured an IVVS probe prototype based on a rad-hard concept and driven by commercial micro-step motors, which has demonstrated satisfying viewing and metrology characteristics at room conditions. The probe sends a laser beam through a reflective prism. By rotating the axes of the prism, the probe can scan all the environment points except those present in a shadow cone. The backscattered light signal is then processed to measure the intensity level (viewing) and the distance from the probe (metrology). During the last years, in order to meet all the ITER environmental conditions, the probe mechanical design was significantly revised introducing a new actuating system based on piezo-ceramic actuators and also improved with a new step focus system. The optical and mechanical schemes have been then modified and refined optimizing the dimensions in order to meet also the geometrical constraints and to integrate the probe with the deployment system. The piezo-motors and the main optical components have been also tested and qualified in ENEA testing facilities reproducing conditions similar to the ITER ones. The paper describes the mechanical concept design solutions adopted in order to fulfill IVVS probe functional performance and taking into account both ITER working environment and strict geometrical constraints.

Id 885

Abstract Final Nr. P1.124

Initial demo Tokamak design configuration

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The realization of a Demonstration Fusion Power Reactor (DEMO) to follow ITER, with the capability of generating several hundred MW of net electricity and operating with a closed fuel-cycle by 2050 is viewed by Europe and many of the nations engaged in the construction of ITER as the remaining crucial step towards the exploitation of fusion power. The recent EU fusion roadmap Horizon 2020 advocates for a pragmatic approach and considers a pulsed "low extrapolation" DEMO. This should be based on mature technologies and reliable regimes of operation, as much as possible extrapolated from the ITER experience, and on the use of materials and technologies adequate for the expected level of neutron fluence. In the new Consortium Eurofusion a corresponding conceptual design activity is being launched. To prepare the DEMO conceptual design phase a number of physics and engineering assessments were carried out in the recent years in the frame of EFDA concluding in an initial design configuration of the DEMO tokamak. This paper gives an insight into the identified engineering constraints and requirements and describes their impact on the selection of the technologies and design principles of the main tokamak components. The EU DEMO program aims at making best use of the technologies developed for ITER (e.g., magnets, vessel, cryostat, and to some degree also divertor and H&CD systems are based on ITER technology). However, other systems in particular the blanket require advanced technologies and design concepts to comply with the main differences between DEMO and ITER. These include the two orders of magnitude larger lifetime neutron fluence, the consequent radiation dose levels, which limit remote maintenance options, the requirement to use low-activation steel for in-vessel components that must operate at high temperature for efficient energy conversion, and the requirement to breed, to extract, to process and to recycle the tritium needed for plasma operation.

Id 108

Abstract Final Nr. P1.125

Methodology to derive the basic DEMO tokamak configuration CAD geometry from system code studies

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To prepare the DEMO conceptual design phase a number of physics and engineering assessments were carried out in the recent years in the frame of EFDA Power Plant Physics and Technology. A number of initial tokamak design options are being studied. The methodology to typically identify and develop the design point of fusion reactor devices involves a variety of tools, including systems codes (e.g. PROCESS), which use simplified (often 0D or 1D) interacting models of plant systems to identify self-consistent plant operating points, and more detailed modelling and engineering analysis to confirm and develop the operating point. This paper describes the procedure that has been developed and applied to derive the radial build, the vertical cross section, and ultimately the 3D configuration model of the tokamak based on the system code output parameters. This procedure reviews the analysis of the radial and vertical build provided by the system code to verify critical integration interfaces, e.g. missing or too large gaps and/or insufficient thickness of components, and update these dimensions based on results of more detailed analyses (e.g. neutronics, plasma scenario modelling, etc.) that were carried out outside of the system code in the past years. As well as providing a basis for integrated engineering analysis, the results can also be used to refine the systems code model. This method, subject to continuous refinement, controls the derivation of the main machine parameters and ensures their coherence vis-à-vis a number of agreed controlled physics and engineering assumptions.

Id 136

Abstract Final Nr. P1.126

Remote handling of material samples and conditioning in a plasma generator operated in a hot cell

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Within the framework of fusion technology new materials for the first wall are investigated. These materials must withstand reactor relevant high neutron fluxes which change the structure and properties of the material and cause radioactivity. The impact of the neutron damage on the plasma-material interaction with respect to thermal and particle loads is investigated in special linear plasma generators in a hot cell. The exposure of probes causes a contamination of the vessel and the build in components itself. In order to avoid an emission of possibly radioactive contamination and to get stable conditioning a real lock system for the probes would be a benefit. A preheating procedure of the probe for cleaning should be possible before the investigation. Minimizing the amount as well as the surface of the required build in components and the installation of cold spots for the condensation of the vaporous contamination might achieve a certain local concentration and waste deposit. Nevertheless all mountings and the operations like the moving and fixing of the probe are to be designed, all without the need of maintenance and revisions but with the assurance of a high operational safety. The intention should be to arrange actuating drives and steerings outside the vacuum as much as possible. Therefore the concept and design of the object holder and the intake is of special importance. Furthermore in case of problems the radioactivate probe should be removed from the vessel in a simple as possible way. Concepts and detailed solutions will be presented.

Id 730

Abstract Final Nr. P1.127

Global shutdown dose rate maps for a DEMO conceptual design

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For the calculations of highly reliable shutdown dose rate (SDR) maps in fusion devices like a DEMO plant, the Rigorous-2-Step (R2S) method is nowadays routinely applied using high-resolution decay gamma sources from initial high-resolution neutron flux meshes activating all materials in the system. The calculation effort to produce global activation and SDR maps is tremendous and is usually tackled by splitting up the geometry for separate R2S calculations. This approach has been utilized in the present paper with the objective to provide SDR relevant for RH systems of a conceptual DEMO design developed in the EU. The primary objective was to assess specific locations of interest for RH equipment inside the vessel and along the extension of maintenance ports. To this end, a provisional DEMO MCNP model has been used, featuring HCLL-type blankets, tungsten/copper divertor, manifolds, vacuum vessel with ports and toroidal field coils. The operational scenario assumed 2.1 GW fusion power and a life-time of 20 years with plant availability of 30%, where removable parts will be extracted after 5.2 years. Results of absorbed dose rate distributions for several relevant materials as well as equivalent dose rate maps are presented and discussed in terms of the different contributions from the various activated components. The importance of different activated regions to SDR values at specific locations vary, depending not only on the physical boundary conditions, but also on computational parameters, most importantly on the mesh spacings of the chosen decay gamma sources. The effect of varying mesh spacings in a full-torus calculation has been studied to assess the impact on specific SDR locations. The paper will conclude with recommendations on best-practice approaches to reliable full-torus SDR calculations.

Id 369

Abstract Final Nr. P1.128

Considerations on Electrical Properties of Liquid Hydrogen and Deuterium

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A precise operation and optimum control of distillation columns in a deuterium and tritium separation plant need a very accurate knowledge of the thermodynamic and transport properties of carried substances. This paper presents a series of equations describing the evolution with temperature of some relevant parameters of hydrogen and deuterium, such as the dielectric permittivity and density, starting from a series of known experimental data. These equations allow an accurate calculation of the various properties of liquid deuterium at different working conditions and indirectly to obtain input data for an improved automatic control system of the isotopic separation columns.

Id 396

Abstract Final Nr. P1.131

Neutronics analysis of water-cooled ceramic breeder blanket for CFETR

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As one of breeding blanket candidates for China Fusion Engineering Test Reactor (CFETR), a water-cooled ceramic breeder blanket with superheated steam is being carried out. In the current design, tritium breeder of lithium titanate (Li2TiO3) pebbles and neutron multiplier of beryllium metallic compounds (Be12Ti) pebbles are mixed in the breeding zones, which are enclosed by structural material of the reduced activation ferritic/martensitic steel (RAFM) and cooled by water of 7 MPa, additionally inserted with several independent multipling zones of elemental beryllium. In this paper, neutron flux profile, archivable tritium breeding ratio (TBR) and nuclear heating distribution were calculated using Monte Carlo code MCNP with the IAEA Fusion Evaluated Nuclear Data Library FENDL3.0 and refined MCNP geometry model of CFETR. Moreover, study on influence of varying neutron spectrum caused by nuclide burn-up within tritium breeding blanket plays a crucial role in the safety, reliability and efficiency of fusion reactors. For this purpose, activation calculation code FISPACT-2007 with EAF-2007 library was applied for computing nuclide inventory so that time dependent neutron spectrum is available for analysis. Finally, lithium invnetory, time related TBR and tritium production could be evaluated through iterative calculation of MCNP and FISPACT.

Id 486

Abstract Final Nr. P1.132

Evaluation on the heat removal capacity of first wall of water-cooled blanket for CFETR

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China Fusion Engineering Test Reactor (CFETR) is an ITER-like superconducting tokamak reactor, which requires tritium self-sufficiency under 200MW fusion power. Three options for blanket design are being carried out, including helium-cooled ceramic breeder (HCCB) blanket, water-cooled ceramic breeder (WCCB) blanket, and helium-cooled lithium lead (SLL/DLL) blanket. As an important component of the blanket, the first wall (FW) removes away most of the blanket structure nuclear heat and high temperature plasma radiation heat flux by coolant (Helium gas or water) flowing through the passage. In general, we assumed the uniform heat flux from plasma chamber on first wall is 0.5MW/m2 for conceptual design of fusion DEMO or power plant reactor. However, With ITER, it has been further discovered that the peak heat flux on first wall is at range of 2-5MW/m2. This will cause the basic question on the feasibility of using helium as the blanket FW coolant for CFETR. Based on WCCB blanket basic design scheme for CFETR, the heat removal capacity of the first wall in different operating conditions is evaluated in this paper, which presents analytical models and data results under different cross-sectional dimension, coolant flow rate, plasma heat flux, coolant pressure by the method called thermal equilibrium numerical simulation analysis. A series of diagrams are given, including the relationship between temperature of the structural material and the coolant flow rate, plasma flux and the coolant temperature, cross sectional dimensions and the maximum temperature of the structural material, the rise of the coolant temperature and the coolant flow rate, and so on. The results are analyzed by being compared with those of He-cooled FW. The optimized scheme on heat removal and water flow characteristics of the FW cooling passage is proposed.

Id 484

Abstract Final Nr. P1.133

Structural design and stress analysis of water-cooled ceramic breeder blanket for CFETR

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As one of the candidate breeding blanket concepts for China Fusion Engineering Test Reactor (CFETR), the water-cooled ceramic breeder blanket is being developed. For water-cooled blanket design, the blanket employs modular scheme. Each module uses RAFM (reduced activation ferritic/martensitic) steel as structural material, mixed pebbles of Li2TiO3 and Be12Ti as breeder and main neutron multiplying material, W as armour of first wall, and arranges independent neutron multiplying layer filled with Be pebbles. Water coolant operates under BWR condition and is heated up into superheated steam in the module. Surrounding plasma, there are 4 inboard modules and 6 outboard modules. Toroidal dimension for each module is allocated at 11.25 degrees. On the basis of radial layout obtained by means of investigation of neutronics performance analysis and preliminary thermal analysis, the structure design of equatorial outboard module is carried out. This module features a steel box enclosed by poloidal U-shaped first wall, side wall, and back plate. By optimization anlysis of structure, box is seperated into 6 sub-modules by the radial-poloidal stiffening plates with internal coolant channels. Each sub-module is strengthened by three toroidal-poloidal stiffenning plates together with radial-poloidal stiffenning plates against in-box LOCA. In order to reduce mass of structural material to improve tritium breeding ratio, the cooling pipes replace the cooling plates and embed in mixted pebble bed zone at some spacing. This paper describes detailed structure of equatorial outboard module and presents stress analyses under normal operation condition and accident condition, using commercial ANSYS code. The module structure is assessed by Structural Design Criteria for ITER In-vessel Components (SDC-IC).

Id 581

Abstract Final Nr. P1.134

Study of impacts on tritium self-sufficiency in a fusion DEMO reactor

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Due to the resource limitation of the tritium availability for future fusion power plant operation in D-T, it is essential to breed tritium to maintain the continuous consumption in the D-T plasma so as to sustain the required fusion power. A minimum criterion of tritium breeding ratio (TBR)>1.05 is commonly required to ensure tritium self-sufficiency, allowing a 5% safety margin to cover loss and retention in the fuel cycle plant. In a DEMO reactor, there are various factors impacting the availability and effectiveness of fusion generated neutrons for production of tritium. They include tritium breeders, neutron multipliers, structural materials and first wall materials, coolants, first wall thickness, and the blanket coverage such as gaps between toroidal and poloidal blanket modules, divertor geometry and the heating and diagnostic ports. Neutron transport calculations were performed based on several selected representative neutronics models for DEMO to quantify the impact of these factors. Overall, without additional neutron multiplier beryllium, liquid breeders can generate more tritium than solid breeders given arbitrary lithium-6 enrichment. Due to competition in neutron interaction, the selected structural material impacts TBR significantly but the detrimental effects can be compensated by enriching the lithium-6 fraction. Tungsten may be acceptable to serve as the first wall as long as the radial thickness is limited to <1 cm. In comparison with Eurofer, 1 cm of tungsten cladding may reduce TBR by ~0.08. The reduction of TBR by a 2cm poloidal gap, compared to no gap between blanket modules, is about 0.008 but the radiation damage due to the streaming paths need to be considered. Allowing space for the divertor or ports may reduce the TBR considerably, by around 0.2.

Id 826

Abstract Final Nr. P1.135

Benchmarking of Monte Carlo tools for nuclear analyses of the European DEMO

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The European Power Plant Physics and Technology (PPP&T) project is devoted to develop a conceptual design of a fusion power demonstration reactor (DEMO). Among the various PPP&T activities a nuclear analysis working package was defined aiming at the development of suitable tools and methodologies for the assessment and optimization of the nuclear performance of DEMO. In the 2012 – 2013 work programmes, a task was dedicated to the evaluation of Monte Carlo (MC) particle transport codes which could replace the well-established MCNP code for fusion neutronics applications to DEMO. In 2012 a preliminary evaluation exercise (bibliography and simplified benchmark case) was carried out, via a collaboration between CEA, CCFE and IPPLM. Several candidates have been identified: TRIPOLI-4, Serpent, FLUKA and Geant4 as promising alternative MC codes. To complete this first evaluation a thermonuclear fusion tokamak type reactor relevant benchmark was performed in 2013 to assess the capability of the selected codes for application to DEMO. Each association has participated using a different MC code: CEA with TRIPOLI-4, CCFE with Serpent and IPPLM with FLUKA. Unfortunately none of the associations has participated with Geant4. This paper aims at present the results obtained in this benchmark exercise on DEMO. The MC code comparisons are based on the MCNP generic model developed by KIT for the so-called DEMO1 baseline. This model does not include blanket modules but a banana shaped blanket container. With the Serpent and FLUKA codes considerable effort was required to translate the MCNP input deck into the syntax of these codes; so limited analyses could only be performed in the frame of the benchmark activity. Regarding TRIPOLI-4, the benchmark has been successfully conducted; the results obtained are statistically similar to those of MCNP with comparable computation times.

Id 525

Abstract Final Nr. P1.136

Module Attachment for DEMO Helium Cooled Blankets

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The European Union is currently developing four different blanket concepts for the future DEMOnstration fusion power reactor that will follow ITER. Two of them, the Helium Cooled Pebble Bed (HCPB) and the Helium Cooled Lithium Lead concepts use helium as primary coolant. Besides the specificities of each blanket concept, the reference DEMO design is based on the Multi Module Segment (MMS) maintenance scheme where each column of blanket modules is attached to a common poloidal bended plate (the Back Supporting Structure – BSS) to avoid direct connection with the Vacuum Vessel (VV) like in ITER. The resulting assembly ("segment") can therefore be removed as a single piece from the top VV ports, hence simplifying remote handling operations. Within the framework of the 2013 Work Program, the European Fusion Development Agreement (EFDA) agency launched a specific task to investigate two different conceptual solutions for the attachment of the modules the BSS, one based on the use of flexible plates welded to the side of the modules and the other on the use of bolted connections on the rear. The two attachment concepts have been developed in collaboration between KIT and CEA, with the requirement to be suitable for both helium cooled blankets. At this preliminary stage of the design, because of the differences in the module manifolds and fluid distribution scheme, their specific implementation is different, but both can be adapted for the two blanket concepts.

Id 366

Abstract Final Nr. P1.138

The design of closed loop regeneration system for molecular sieve dryer in detritiation system

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The combination of hydrogen isotope oxidation reactor and water vapor dryer has been applied to the detritiation system,such as atmosphere detritiation system and vent detritiation system in tritium process facility.Molecular sieve is usualy used as the adsorbent of tritiated water vapor in dryer. After reaching breakthrough,the molecular sieve bed(MSB) needs to be regenerated at about 300º with the purge gas stream in counter current.In order to minimize the release of tritiated water to the enviroment,the purge gas generally flows through a condenser backing up with a cold trap to decrease the water vapor concentration as low as possible. In the present work, a more convenient regeneration method was designed,which is a closed loop regeneration system including a blower,a condenser and two auxiliary small size MSB in parallel with alternation mode—one for work,another for regeneration.The most significant feature is that external purge gas and cold trap are not needed.With the driving force of blower,the eluting gas from working MSB passes through the regenerating MSB,condeser and the auxiliary MSB successively, water vapor in the gas was condensated in condenser cooled by chilling water and adsorbed in auxiliary MSB,then the dehydrated gas went back to the outlet of working MSB.Liquid water in condeser will be storaged in tank. In order to verify the feasibility of this design,the regenaration system was used in a water vapor absorption expriment lasting 30 days,with the dryer molecular sieve packing mass 50kg.The result showed that the regernaraton system is effective for MSB regeneration:the water vapor concentration in the outlet of working MSB was blow 1ppm all the time, the size of auxiliary MSB is less than 1/50 of the working MSB and no water vapor was sent out of the dehydration system.This design seems practical to be used in the detritiation system for its inherent safety and convenience.

Id 670

Abstract Final Nr. P1.139

Hydrogen extraction characteristics of a high-temperature proton conductor for tritium monitoring in fusion fuel cycle and breeding blankets

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A new tritium monitoring system which combined a high-temperature proton conductor (HTPC) with a high-sensitivity proportional counter has been proposed. The high-temperature proton conductor was used as the membrane separator for concentrating tritium, and then the concentrating tritium was detected by the proportional counter. The new tritium monitoring system with measuring radioactive tritium concentration down to 5E-4Bq/cm3 would be used for monitoring low levels of tritium in fuel cycle and breeding blankets of current fusion test tokamaks and future fusion reactors. To evaluate application of the new tritium monitoring system, hydrogen extraction characteristics of a one-end closed tube made of proton-conducing ceramic CaZr0.9In0.1$O_{3-\alpha}$ (O.D.20mm, I.D.17mm, effective electrode area 160cm2) were studied over the temperature range from 873K to 1073K. In the case of helium with 100ppm H2 at 1023K and 0.85V DC, the hydrogen recovery efficiency was more than 99%, suggesting that the extraction of hydrogen could be operated with a current efficiency close to unity. In the electrolysis of 6100ppm water vapor balanced with helium fed to anode at a rate of 100ml/min, hydrogen evolution rate reached maximally 0.28ml/min and the hydrogen recovery efficiency was 46% at 3.01V DC and 1073K, and hydrogen evolution rate decreased to 0.06ml/min with anodic feeding flow decreasing to 10ml/min, but the hydrogen recovery efficiency increased to 97.8%. The principal results indicated that hydrogen can be extraced from anode to cathode through proton conductor by applying DC potential difference, and the hydrogen could be concentrated. For realization the new tritium proposed monitors, extraction characteristics of hydrogen isotopes using very low concentration tritium were also evaluated in this study, HT could be extraced and concentrated from very low concentration tritium, and a larger electrode area would be needed to increase hydrogen isotopes evolution rate and hydrogen recovery efficiency.

Id 994

Abstract Final Nr. P1.140

Conceptual Study on Transmutation Reactor Based on Tokamak Neutron Source

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Concptual study on the transmutation reactor based on tokamak as a neutron source are performed and transmutation characteristics of transuranics (TRU) are investigated. Optimum radial build of a transmutation reactor is found by coupled analysis of the tokamak systems and the neutron transport. The dependence of the transmutation characteristics on an aspect ratio, A in the range of 1.5 to 4.0, and on the fusion power in the range of 100 MW to 500 MW are investigated. Equilibrium fuel cycle is developed for effective transmutation and it is shown that with one unit of the transmutation reactor based on the tokamak producing fusion power in the range of a few hundred MW, up to 10 PWRs (1.0 GWe capacity) can be supported with the portion of Pu and minor actinides properly adjusted.

Id 324

Abstract Final Nr. P1.142

Thermally induced outdiffusion studies of deuterium in ceramic breeder blanket materials after radiation

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Based on a KIT-CIEMAT collaboration on the radiation damage effects of light ions absorption/desorption in ceramic breeder materials, candidate materials for the ITER EU TBM were tested for their outgassing behaviour as a function of temperature and radiation. Lithium orthosilicate based pebbles with different metatitanate contents (from KIT), fabricated by a melt-based process, and pellets of the individual oxide components (from CIEMAT), fabricated by a solid-state process, were dehydrated and subsequently exposed to a deuterium atmosphere at room temperature inside a capsule. Then the thermally induced release of deuterium gas was registered up to 800 °C. This as-received behaviour was studied in comparison with that after submitting the deuterium-exposed samples to gamma irradiation for several weeks. The thermal desorption spectra reveal the different degree of deuterium absorption/desorption behaviour depending on the composition, the porosity and the induced damage. While lithium metatitanate samples exhibit a monotonic spectra centred around 300 °C, the pure silicate composition shows a deuterium-release occurring with several peaks, the highest around 200 °C. In any case, the total outgassing is achieved at temperatures above 500 °C. The results of pellets with individual compositions are discussed and compared with the R-NRA profiles registered on deuterium-implanted samples. The two different experiments represent the basis for a detailed discussion on the thermal behaviour of deuterium in breeder blanket ceramics.

Id 790

Abstract Final Nr. P1.143

Preliminary RAMI analysis of DEMO WCLL blanket and breeder systems

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Demonstration Power Plant (DEMO) will be a prototype fusion reactor designed to prove the capability to produce electrical power in a commercially acceptable way. One of the key factors in that endeavour is the achievement of certain level of plant availability. Therefore, RAMI (Reliability, Availability, Maintainability and Inspectability) will be a key element in the engineering development of DEMO. Although DEMO is currently within conceptual design phase, some studies have been started so as to assess different design alternatives from RAMI point of view. The main objective of these studies is to evaluate the influence on DEMO availability and to focus the critical parts that should be further researched and improved in order to develop a high-availability oriented DEMO design. In this framework a preliminary RAMI analysis of the Water Cooled Lithium-Lead (WCLL) blanket and breeder concept for DEMO has been done. The systems configuration has been defined based on the reference design documentation and on several design assumption and hypothesis. The Plant Breakdown Structure (PBS) and Functional Breakdown Structure (FBS) have been determined for each system. On this basis, Failure Mode and Effect Analysis (FMEA) analysis has been performed. Finally, preliminary Reliability Block Diagram (RBD) analysis has given first reliability and availability results and has highlighted the critical parts and parameters influencing the system availability. Sensitivity analysis has shown the influence of variations of failure rates of critical events and associated mean down times (MDT) on the system availability, and what levels of availability could be reached according to such variations. This paper presents the main outcomes of this analysis, summarizing the results obtained and the main conclusions drawn.

Id 801

Abstract Final Nr. P1.145

Modelling tritium release data from LIBRETTO-4/-5 neutron irradiation experiments

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For more than two decades, LIBRETTO (LIquid BReeder Experiment with Tritium Transport Option) irradiation Programme at HFR-Petten (NL) represents a valuable source of tritium transport data under irradiation for liquid-metal (Pb15.7(2)Li; LLE) blanket materials. LIBRETTO-1 (1989/1990) provided direct demonstration of in-situ breeding and release from closed capsules at reference design temperature ranges (300-500 C). LIBRETTO-2 (closed FM/LLE capsules) (1991-1992) qualified originally PC-Al2O3 in-contact with lead-lithium tritium permeation barrier under n-irradiation providing a preliminary release-rate physical understanding and evidences of He-bubbles formation in lead-lithium eutectic. LIBRETTO-3 (bubbled-up FM/LLE open-capsules) (1993-1995) qualified diverse types of inner tritium permeation barriers (PC-Al2O3, CVD-Al2O3, TiC+Al2O3) and assessed a value of solved tritium mass-transfer coefficient to injected bubbles. The recent LIBRETTO-4 (2005-2010) and LIBRETTO-5 (2009-2010) resume the best historical experimental experience on complex tritium breeding/release in irradiation tests. In LIBRETTO-4 two identical capsules (500–550 C and 300–350 C) tested tritium permeation rate through pre-oxidized FM EUROFER97 into a He + 1000H2 vppm cooling channel. A similar second channel recovered the tritium desorbed from the LLE and permeated through a 316L internal wall. In similar configurations, LIBRETTO-5 temperature range is set up 300-500 C. Direct data analysis of month-cycles has served to select cycles for physical modeling in order to match measurement uncertainties. A numerical release-rate model has been developed to simulate tritium fluxes through capsule materials all along variable experimental conditions. The modeling provides an experimental fitting of long-cycles release curves and permits a coherent physical understanding of release rate mechanisms. Consistent tritium transport data [diffusivity, solubility, mass-exchange coefficients, recombination constant] under irradiation in capsule's material have been derived and are proposed. Role and impact of native bred helium bubbles is discussed.

Id 995

Abstract Final Nr. P1.146

Development of an HCLL-TBM Configuration and Ancillary Systems Dynamic Transfer Model using EcosimPro®

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EcosimPro® is a powerful mathematical tool capable of modeling any kind of dynamic system represented by differential-algebraic equations (DAE) or ordinary-differential equations (ODE) and discrete events. It is an object-oriented non-casual tool that provides an intuitive environment for creating reusable components and making simulations easy. Born in the frame of ESA aerospace program and QA certified for use in a range of fields, EcosimPro is developed by the company Empresarios Agrupados. This simulation tool has already been tested for the simulation of tritium diffusion process by means of a validation process against tmap7, theoretical and experimental results, standing out as a highly useful tool with a great future in modelling and simulating the systems involved in the production and recovery of tritium. One of the key issues in ITER is the strict control of tritium inventory that has to be accomplished throughout the whole process. In particular, Test Blanket Modules (TBM) and their auxiliary systems have a high tritium concentration and this inventory control becomes essential for the development of fusion energy. In this context, a set of EcosimPro libraries called TRITIUM_LIBS has been developed for simulating tritium transfer phenomena in the Test Blanket System (TBS) in order to determine the amount of tritium per unit time that can be collected, accounted for and routed out of the TBS auxiliary systems to the ITER Tritium Plant. Simulation models of the following Tritium plant sub-systems have been developed: Helium Cooled Lithium Lead (HCLL) TBM, LiPb loop: Tritium Extraction Unit (TEU), Tritium Removal System (TRS), Helium Coolant System (HCS), Coolant Purification System (CPS) and Auxiliary units. Those simulation models enable to perform parametric analysis at different operating conditions of the TBM ancillary systems under various ITER irradiation scenarios and at different TBS dimensions and materials, as shown at the ISFNT-11.

Id 505

Abstract Final Nr. P1.147

Model improvements for tritium transport in DEMO Fuel Cycle

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In normal working conditions, the DEMO operation requires a large amount of tritium which is directly produced inside the reactor by means of Li-based breeder blanket before being then recovered and purified through dedicated systems. In this framework, tritium is in contact with large surfaces of hot metallic walls and, therefore, it can easily permeate through the blanket cooling structure, reach the steam generator and finally the environment. For these reasons, the development of dedicated simulation tools able to predict tritium losses and inventories are necessary to verify the accomplishment of the commonly accepted tritium environmental releases. In the last two years, the FUS-TPC code was developed to estimate tritium inventories and losses in three different blanket concepts considered for DEMO (i.e. HCPB, HCLL, WCLL). Via a simplified T migration model throughout the overall T cycle, the FUS-TPC code allowed to highlight the main design requirements, mainly in terms of permeation control techniques under steady state operation. In this work, the FUS-TPC code was lately improved by including a unified surface/diffusion permeation model and the possibility to choose between steady state vs. pulsed regime. Moreover, this activity introduces the tritium generation profile along the blanket radial coordinate by arranging a novel 1-D model. The model was applied to the inboard segment on the equatorial axis of the HCLL DEMO blanket of length about 0.5 m. The profiles of tritium partial pressure in Li-Pb in the cooling and stiffening plates were obtained by assuming several PRF (Permeation Reduction Factor) values. In particular, assuming a the tritium extraction system efficiency of 0.8, along the radial axis of the considered blanket segment the tritium partial pressure moves from 0.3 to 7.7 Pa. Future improvements will consider the application of the model to all segments of different blanket concepts.

Id 307

Abstract Final Nr. P1.148

Experimental validation of water gas shift and isotopic swamping reactions for highly tritiated water decontamination via Pd-membrane reactor

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The design of the fuel cycle in the next D-T fusion machines, like ITER and DEMO has to consider the processing of hydrogen isotopes in several chemical varieties (elemental form, oxides, hydride, hydrocarbons, etc.). In this scenario an open issue is represented by the treatment of Highly Tritiated Water (HTW) formed mainly during cryopump regeneration and tritium release into a glove box. The tritium concentration in such HTW is above 1013 Bq/kg and, therefore, it cannot be processed in the water detritiation system and, for safety reason, its long term storage has to be avoided. The treatment of 2 kg of stoichiometric HTW containing up to 300 g of tritium, generated by an ITER-like machine, has been previously simulated by a numerical code in order to evaluate the detritiation capability of a Pd-based membrane reactor through two different processes: the Water Gas Shift (WGS) and the Isotopic Swamping (IS). In this work the simulation code is validated by dedicated experiments. A membrane reactor consisting of a stainless steel shell hosting a Pd-Ag permeator tube (wall thickness 0.15-0.20 mm, lenght 500 mm, diameter 10 mm) is tested under the operating conditions assumed in the simulation analysis. Particularly, the Pd-reactor is tested in temperature range 573-673 K, lumen pressure range of 100-400 kPa and protium sweep flow rate in the shell side between 50 and 150 Ncm3min-1. A particular attention is dedicated to the efficiency and selectivity of catalyst. In fact, the results of the simulation show that the detritiation performance of the WGS process is higher compared to the IS one. However, a practical weak point of WGS process to be experimentally verified is represented by the presence of side reactions that may produce tritiated species, particularly methane. Therefore, the experiments are especially dedicated to test novel catalysts able to minimize methane formation.

Id 877

Abstract Final Nr. P1.149

Electro-Magnetic Analysis of the European Test Blanket Modules for ITER

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The European Domestic Agency for ITER (Fusion For Energy, F4E) is currently developing two Test Blanket Modules (TBMs): the Helium-Cooled Lithium-Lead (HCLL) and the Helium-Cooled Pebble-Bed (HCPB) concepts. From the electromagnetic point of view the TBM is a very complex system. In fact it is exposed to an intense magnetic field produced by the superconducting coils, it is made of electrically conductive and ferromagnetic structures on which electromagnetic transients induce eddy currents circulation. For this reason a careful electromagnetic assessment is required for assuring reliable operations as well as its mechanical integrity. In this paper we present the results of several transient electromagnetic analyses for determining corresponding loads (torques and net forces) due to Lorentz and Maxwell (in the ferromagnetic regions) forces computed via the Maxwell stress tensor method. In particular major disruptions (MDs), vertical displacement events (VDEs), magnet fast discharges (MFDs) and entire ITER operational scenarios are analyzed. The three-dimensional finite element (FE) model used for all the analyses comprises the two (HCLL and HCPB) TBMs including the shield, frame, port plug, pipes and all the electrical connections among these components. It includes also the neighboring regular blanket modules around the port, the Vacuum Vessel (VV) sector with port structure and ferromagnetic insert, the divertor electrically connected to the VV, the triangular support with its copper cladding. The material properties take into account the nonlinear magnetization curve of the EUROFER and the dependence of its electrical resistivity upon temperature. Cyclic symmetry boundary conditions and the ANSYS electromagnetic edge-based formulation are used. In the paper we describe the techniques used for simulating the electromagnetic phenomena involved and the postprocessing of the results to obtain the loads acting on the structures. Finally we summarize the loads applied to the TBMs different components and give a critical view of the results.

Id 267

Abstract Final Nr. P1.150

Detailed Design of the ITER Torus Cryopumps Cold Valve Boxes

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Fusion for Energy is responsible for the in-kind supply of 6 torus, 2 cryostat, 3 heating and 1 diagnostic neutral beam cryopumps and their 12 Cold Valve Boxes (CVBs). The cryolines and Warm Regeneration System are also part of the Fusion for Energy supply. The CVBs are key components to control the necessary supply and return of helium to the pumps in order to guarantee the ‘on-demand’ operation and regeneration requirements of all cryopumps. They also provide the necessary valves for the safe operation of these pumps. They have been designed in close collaboration between F4E, ITER IO and IDOM. The Torus Cryopump CVB consists of a complex network of pipes and valves with operating temperatures ranging from 4.5K to 470K. With a vacuum vessel of 1.8m diameter and 2.3m height, this CVB features 25 control valves, 15 pressure relief valves, 8 Johnston couplings, a thermal shield, and a collector header, as well as interfaces to the cryoline, warm regeneration lines, cryogenic guard vacuum system, the cryopumps and other clients. The total weight of the CVB is approximately 6 tons. The detailed design of the Torus Cryopump CVB has been finalised and accounted for the P&ID definition, component and especially pipe and valve (control and safety) sizing, structural analyses of the complete CVB model in ANSYS (including analyses with CAESAR II for piping and SAP2000 for the support frame), validation according to EN Code and final detailed 3D model in CATIA. The paper will demonstrate that the design of the CVB satisfied the highly demanding ITER operational and accident conditions, pressure drop and heat loss limitations, and safeguarded the performance of components, especially those with a safety function during extreme (seismic and accident) events inside an extremely tight environment.

Id 287

Abstract Final Nr. P1.151

Tritium permeation in the presence of hydrogen

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The permeation of a single hydrogen isotope has two distinct limiting transport regimes: one diffusion-limited, in which the permeating flux varies with the square root of the partial pressure, and the other surface-limited, occurring at lower partial pressures, in which the dependence is linear [1]. For the case of two isotopes, a third regime has been discussed [2] in which the tritium permeation varies linearly with HT pressure in the presence of hydrogen, under the assumption that transport is diffusion-limited. Furthermore, it has been shown [1] that for a single isotope the transition from surface- to diffusion-limited transport is governed by a dimensionless "permeation number" that depends on the material properties and the square root of the partial pressure. A generalization of the permeation number based on the total pressure of all hydrogen isotopes has been suggested in [3]. In this work we outline a general formulation for multiple isotope permeation, and demonstrate how the three transport regimes mentioned above arise as limiting cases delineated by two dimensionless groups, the isotope pressure ratio and the generalized permeation number given in [3]. In the diffusion-limited case, addition of hydrogen can effectively suppress tritium permeation [2], and it is shown here how this regime can be easily identified using the aforementioned dimensionless groups. Some implications related to permeation experiments and the design of systems such as ceramic breeder blankets (which typically include hydrogen in the purge gas) are outlined.
[1] I. Ali-Khan, K. J. Dietz, F. G. Waelbroeck, and P. Wienhold, J. Nucl. Mat. 76/77 (1978) 337-343. [2] J. T. Bell and J. D. Redman, J. Phys. Chem. 82 (26) (1978) 2834-2838. [3] D. F. Holland and G. R. Longhurst, Fus. Tech.8 (1985) 2067-2073.

Id 441

Abstract Final Nr. P1.152

Features of structural-phase states and mechanical properties of vanadium alloys subject to thermomechanical treatment modes

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The RF goal of the material science R&D is the manufacture of the heat strength and corrosion resistant low activation structural materials for the temperature window (300 – 850 °C) and radiation load up to 160 dpa-Fe. Vanadium alloys have good prospects for such applications. The required properties are achieved by methods of combined element compositions and different thermo-mechanical treatment (TMT) modes of vanadium alloys. The regularities of heterophase structure formation and mechanical properties of low-activated vanadium alloys of systems V-Ti-Cr-C-O-N (alloy V-4.21Ti-4.36Cr-0.013C-0.02O-0.011N), V-Zr-C (alloy V–2.4 Zr–0.25 C–0.04 O–0.01 N) and V-Me(Cr, W)-Zr-C (alloys V-8.75Cr-0.14W-1.17Zr-0.01C-0.02O-0.01N and V-4.23Cr-7.56W-1.69Zr-0.02C-0.02O-0.01N) (wt.%) after different TMT regimes were investigated. TMT regimes that provide the possibility of transformation of coarse particles of initial metastable vanadium carbide into uniformly distributed nano-sized particles of stable non-metallic phases were developed. The TMT regimes increase the thermal stability of highly-dispersed heterophase state, the collective recrystallization temperature and the short-term strength of the alloys. It is shown that alloys of the V-Cr-W-Zr-C-O-system are prospective for further R&D of advance vanadium alloys and methods of increasing their heat strength resistance while reducing the tendency to cold brittleness. In these alloys a significant increase in short-term high-temperature strength (at 800 °C) is achieved while maintaining high reserve of ductility at room temperature. Opportunities of further increase in heat strength resistance of vanadium alloys are associated with the variation (increasing) of volume fraction of non-metallic phase by controlled modification of the alloy composition by carbon and oxygen.

Id 892

Abstract Final Nr. P1.153

Tritium release behavior from neutron irradiated beryllium

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Tritium release behavior from neutron irradiated beryllium Igor Kupriyanov 1, Andrey Markin 2 1 A.A. Bochvar High-Technological Research Institute of Inorganic Materials (VNIINM), Moscow, Russia 2 Institute of Physical Chemistry of Russian Academy of Science, Moscow, Russia The efficiency of beryllium for fusion applications will be strongly depended on its behavior under neutron irradiation. The most important consequences of neutron irradiation of beryllium are helium induced swelling and tritium retention and release. The effect of neutron irradiation on tritium release from beryllium is described in this paper. Beryllium samples were irradiated in the SM reactor with neutron fluence (E 0.1 MeV) of 2 x 1022 cm-2 at 200oC. Mass-spectrometry technique was used in out of pile monitoring of tritium and helium releases during linear temperature ramping within a temperature range from RT to 1300oC. The first signs of tritium release in the form of HT molecules were detected at temperature of 300oC. Then, at temperature of 400oC the release of T-2 molecules also began. The maximal release rate of T-2 molecules was fixed within the temperature range 900-1000oC. With further increase of temperature, the release rate decreased and reduced up to a background level at temperatures still below the melting point. The total amount of tritium released as T-2 molecules was 60 appm. The helium release started in small amount at 300 oC, when first mass 4 was detected. The main part of helium was released in the temperature range from 1200 oC to melting point. The total amount of helium released from irradiated beryllium was 4300 appm.

Id 881

Abstract Final Nr. P1.154

Hydrogen uptake from plasma and its effect on EUROFER 97 and ODS-EUROFER steels at elevated temperatures

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Hydrogen effect on the mechanical properties of the ferrite-martensite EUROFER 97 and ODS-EUROFER steels was studied under continuous hydrogen charging from hydrogen-enriched plasma at temperatures up to 400 oC. Hydrogen uptake measured with TDS method was found to be markedly higher in ODS-EUROFER steel in comparison to that in EUROFER 97 steel evidencing on high hydrogen trapping ability of the oxide-dispersion strengthening yttria nanoparticles. The hydrogen uptake was measured depending on charging temperature and applied electrical potential between the studied steel specimen and plasma of glow discharge. It is found that hydrogen has only a minor effect on the yield stress and tensile strength of the studied steels over the whole range of testing temperature. EUROFER 97 and ODS-EUROFER steels manifest rather different sensitivity to hydrogen embrittlement with increase of the applied electrical potential. It is shown that hydrogen reduces markedly elongation to fracture of ODS-EUROFER steel, if hydrogen concentration exceeds a certain critical value. FEG-SEM fractography and TEM observations were performed in terms of investigation of the hydrogen-induced fracture mode at elevated testing temperature. Possible role of the interface between yttria nanoparticles and steel matrix and the distribution yttria nanoparticles in the hydrogen embrittlement mechanism are discussed.

Id 296

Abstract Final Nr. P1.155

Laser welding with hot wire technology applied on the austenitic stainless steel of ITER Correction Coil Case

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ITER Correction coil case reinforces the winding packs against the electromagnetic loads , minimizes stresses and deformations of winding pack. The cases are made of high strength and high toughness austenitic stainless steel (316LN) hot rolled heavy plate and has a thickness of 20 mm. Considering the small cross section and large dimensions of the case, deformation of the case closure welding become a challenge to the traditional welding method. The laser welding, as the advanced welding technology, provides feasibility of the case closure welding. In this paper, the multi pass laser welding with hot wire technology was used, researching the laser weldability of 20 mm thickness 316LN austenitic stainless steel and analysing the microstructure of the welded joint. The welding experiments were YLS-4000 fiber laser (IPG) and TS5000 hot wire feeder system (Fronius), the high-Mn content weld filler 316LMn was used for the welding process. The result shows the welded joint, which testded by radiographic and penetration testing, was no obvious surface and internal defects based on the optimized welding parameter. The welded joint was fully austenitic microstructure and displayed of cellulars and columnar grain. The grain size of the heat affected zone (HAZ) was not growing clearly. The microhardness of the fusion metal and HAZ was higher than base metal without apparent softening zone.

Id 382

Abstract Final Nr. P1.156

Elaboration and thermomechanical characterization of W/Cu functionally graded materials produced by Spark Plasma Sintering for plasma facing components

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In future fusion reactors, such as ITER, one of the key components is the Divertor. It is essentially divided in two parts: a support structure called cassette body and plasma facing components (PFCs) devoted to sustain heat flux in the range of 10 MW.m-2 in steady state and up to 20 MW.m-2 during transient phases. PFCs consist of the dome, inner and outer vertical targets. So far monoblock technology has been retained for these vertical targets. They are made of armor material (tungsten W) and structural material (CuCrZr). Due to thermal expansion coefficient difference, an interlayer of pure copper is used as compliant layer between the armor material and the CuCrZr heat sink. A promising behavior in terms of thermal fatigue lifetime under heat flux up to 15 MW.m-2 has been shown for the W-armored components. But for higher heat fluxes, the W-embrittlement as well as the failure issues between the Cu-heat sink and the W-armored, under thermal cycling conditions at high temperature, remains a technologic challenge to guarantee a safe operation of the W-armored components. In order to reduce the risk of bonding failures, it is proposed to replace the soft layer by a continuous functional graded material (FGM) of Cu/W. These types of materials avoid interfaces and may improve the thermo-mechanical resistance of the components. In this work, we investigate the fabrication and the thermo-mechanical behavior of some W/Cu@FGM configurations prepared by SPS processing. These configurations were chosen with regard to their optimized thermo-mechanical behavior which was evaluated with FEM simulations [1].To manufacture adequate FGM samples, SPS parameters were optimized independently for W powder, and Wx/Cu1-X powder mixtures. Structural properties, thermal conductivity and tensile stress of sintered Wx/Cu1-X powder were evaluated. Secondly, W/Cu@FGM configurations have been prepared taking into account optimized SPS parameters of individual materials. Thermo-mechanical properties of these FGM materials were assessed. [1] E. Autissier et al, Fus.EngDes. 88.9-10 (2013): 1714-1717

Id 438

Abstract Final Nr. P1.157

Manufacturing of self-passivating tungsten based alloys by different powder metallurgical routes

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Tungsten is presently the main candidate material for the first wall armour of future fusion reactors. However, the use of tungsten represents a serious safety concern in case of a loss-of-coolant accident with air ingress into the reactor vessel. In such a situation, the high temperatures achieved in the in-vessel components due to the decay heat would lead to fast tungsten oxidation with the release of volatile radioactively activated tungsten oxides. A possible way to prevent tungsten oxidation is the addition of oxide forming alloying elements that form a self-passivating layer at high temperatures in presence of oxygen. In previous works, different ternary tungsten alloys of the systems W-Cr-Si and W-Cr-Ti were manufactured by mechanical alloying (MA), compaction, glass encapsulation and densification by Hot Isostatic Pressing (HIP). The obtained alloys exhibited nearly 100% density, extremely fine and homogeneous microstructure with grain size below 500 nm and an oxidation rate several orders of magnitude lower than pure tungsten at temperatures up to 1000°C due to the formation of a protective oxide layer. However, the glass encapsulation process involves greater difficulties for large ingots than in the case of small samples, as well as the risk of poor resistance of the capsule during HIP, preventing full densification. In view of a future upscaling to larger sizes, different powder metallurgical routes are being explored for the manufacturing of self-passivating tungsten alloys. In this work, the properties of self-passivating tungsten alloys of the system W-Cr-Ti manufactured by MA, compaction, pressureless sintering in H2 and subsequent HIPing without encapsulation are compared to those obtained by HIPing of mechanically alloyed powder encapsulated in metal canisters, as well as to samples produced by the previous route of MA-compaction-glass encapsulation-HIP. Microstructure, mechanical properties and oxidation resistance of samples produced by the different routes are compared.

Id 974

Abstract Final Nr. P1.158

Microstructural characterization of ODS ferritic steels at different processing stages

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Nanostructured Oxide Dispersion Strengthened reduced activation ferritic stainless steels (ODS FS) are promising materials for structural applications in fusion power reactors. ODS FS show high creep strength, reasonable fracture toughness and improved radiation damage resistance at the temperature of operation, up to about 750 °C, due to the presence of a dispersion of Y-Ti-O nanoclusters and an ultrafine microstructure. Ferritic stainless steel powders were produced by gas atomization and HIPped after adjusting their oxygen, Y and Ti contents to the required value to form Y-Ti-O nanoclusters during subsequent heat treatments. These nanoclusters would be embedded in a ferritic stainless steel matrix (Fe - 14 Cr – 2 W). HIP consolidation was performed at different temperatures, in order to study the microstructural changes of the ferritic phase, the evolution of the Cr-, Y- and Ti-rich oxides and the precipitation of nanoclusters. Post-HIP heat treatments were carried out at relatively high temperatures (1270 and 1300 °C) to evaluate the feasibility of achieving both a complete dissolution of the oxides on prior particles boundaries and an ultrafine Ti- and Y-rich oxides precipitation. Final thermo-mechanical treatments were carried out in order to obtain an ultrafine ferritic microstructure and to break any possible oxide network. FEG-SEM and TEM were used to characterize the microstructure of the atomized powders and the ODS FS specimens after HIP consolidation, post-HIP heat treatments and thermo-mechanical treatments.

Id 990

Abstract Final Nr. P1.159

Advanced Examination Techniques applied to the Qualification of Critical Welds for the ITER Correction Coils

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The ITER Correction Coils (CCs) consist of three sets of six coils located in between the toroidal (TF) and poloidal field (PF) magnets. The CCs rely on a Cable-in-Conduit Conductor (CICC), whose supercritical cooling at 4.5 K is provided by helium inlets and outlets. The assembly of the nozzles to the stainless steel conductor conduit includes fillet welds requiring full penetration through the thickness of the nozzle. Static and cyclic stresses have to be sustained by the inlet welds during operation. Severe constraints are imposed to these welds, both in terms of position (welding on vertical conductor) and of maximum temperature allowed on the superconducting strands during welding. Moreover, the welds are submitted to the most stringent quality levels of imperfections according to standards in force. The entire volume of helium inlet and outlet welds is virtually uninspectable with sufficient resolution by conventional or computed radiography or by ultrasonic testing. On the other hand, X-Ray Computed Tomography (CT) was successfully applied to inspect the full weld volume of several dozens of helium inlet qualification samples. For CC welds, the technique is applied in laminographic “shear” mode that allows exploring, layer after layer, the entire volume of the welds. Weld imperfections, including the thinnest planar defects, can be exhaustively imaged and their position precisely determined in the volume. Defects can finally be quantified and assessed according to specified quality levels. Destructive tests carried out on qualification welds have demonstrated in addition an impressive one-to-one correspondence between the individual laminographic cuts and micro-optical observations. The extensive use of CT techniques allowed a significant progress in the weld quality of the CC inlets. CT is also a promising technique for inspection of qualification welds of helium inlets of the TF magnets, by far more complex to examine due to their larger dimensions.

Id 160

Abstract Final Nr. P1.160

Extensive characterisation of advanced manufacturing solutions for the ITER central solenoid pre-compression system

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The ITER Central Solenoid (CS), positioned in the center of the ITER tokamak, will provide a magnetic field, contributing to the confinement of the plasma. The 13 meter high CS consists of a vertical stack of 6 independently driven modules, dynamically activated. Resulting opposing currents can lead to high separation forces. A pre-compression structure is implemented to counteract these opposing forces, by realising a continuous 180 MN coil-to-coil contact loading. Preload is applied by mechanical fastening via 9 subunits, positioned along the coil stack, each consisting of 2 outer and 1 inner tie plate. The tie plates therefore need to feature outstanding mechanical behavior in a large temperature range. High strength, Nitronic®-50 (F XM-19) austenitic stainless steel is selected as candidate material. The linearised stress distribution reaches approximately 250 MPa, leading to a required yield strength of 380 MPa at room temperature. Two different manufacturing methods are being studied for the procurement of these 15 meter long tie plates. A welded solution originates from individual head- and slab-forgings, welded together by Gas Metal Arc Welding (GMAW). In parallel, a single piece forged solution is proven feasible, impressively forged in one piece by applying successive open die forging steps, followed by final machining. Maximum internal stress is experienced during cool-down to 4 K as a result of a large discrepancy in thermal contraction between the support system and the coils. Furthermore, the varying magnetic fields in the independently driven coils introduce cyclic loading. Therefore, assessment of the two manufacturing solutions, in terms of both static and dynamic mechanical behavior, is performed at ambient as well as cryogenic temperature. An extensive characterisation including non-destructive, microstructural and mechanical examination is conducted, evaluating the comparative performance of both solutions, reporting, amongst others, yield strength reaching the requirement for both solutions.

Id 322

Abstract Final Nr. P1.161

Displacement damage effect on the radiation induced deuterium absorption for different types of SiC

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SiC materials are primary candidates for flow channel inserts in blankets due to their excellent thermo-mechanical and corrosion properties. For this application hydrogen isotope absorption is of fundamental importance. During reactor operation the SiC material will be exposed to tritium in a hostile radiation environment. Absorption, diffusion, and desorption are expected to occur, depending strongly on the radiation conditions, neutron flux, and ionizing radiation. The main aims of this work are to compare different types of SiC in terms of ionizing radiation induced deuterium absorption, and to address displacement damage effects. Reaction bonded, hot pressed, physical vapour deposited, and chemical vapour deposited SiC samples were deuterium loaded for different times at 1.2 bar. Loading was carried out either without irradiation, or with both sample and surrounding deuterium gas irradiated with 1.8 MeV electrons to evaluate the radiation enhanced deuterium absorption. After deuterium loading, with or without irradiation, the samples were heated up to 800 C and the deuterium released was measured as a function of temperature. The amount of deuterium absorbed during irradiation strongly depended on the type of SiC material. In order to evaluate displacement damage effects on the ionizing radiation induced deuterium absorption, samples of the 4 materials were subjected to 45 keV neon implantation to produce damage in the surface region, and then deuterium loaded during 1.8 MeV electron irradiation. It was observed that the deuterium absorbed increased noticeably due to the displacement damage produced in the different SiC materials. Furthermore, while for the unimplanted samples the deuterium amount absorbed during irradiation was very different for each material type, after implantation / damage the deuterium absorbed became very similar irrespectively of the type of SiC.

Id 400

Abstract Final Nr. P1.162

Radiation induced deuterium absorption dependence on irradiation 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. Samples of various types of SiC have been irradiated making use of a special chamber with a 50 µm 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 indicate that the radiation induced deuterium absorption does not depend on dose rate but depends linearly on total dose. The amount of absorbed deuterium depends linearly on deuterium gas pressure. The behaviour with irradiation temperature is more complex, and clear changes in the deuterium thermal desorption are observed to occur depending on irradiation temperature.

Id 516

Abstract Final Nr. P1.163

Oxidation recovery of radiation induced surface damage in aluminas: luminescence qualification

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In future fusion devices, oxide ceramics such as Al2O3 for insulation applications will be subjected not only to neutron and gamma irradiation, but also bombardment by H isotope and He ions accelerated by local electric fields. This produces an increase of surface electrical conductivity, potentially as serious as volume degradation. The degradation has been shown to be due to the progressive formation of oxygen vacancies (F and F+ centres), vacancy aggregates, and aluminium colloids in the surface region, enhanced by oxygen loss caused by preferential sputtering during irradiation. Associated ion beam induced luminescence (IBIL) being developed as a tool to monitor material modification, indicates that partial recovery can be achieved by exposure to air (reoxidation). Initial attempts to reverse the degradation by reoxidation produced an apparent full recovery of surface conductivity, but only partial recovery of the luminescence, due to the larger material depth contributing to the IBIL, compared with the surface conductivity. Work has now explored the possibility of post irradiation recovery by prolonged thermal annealing in air and oxygen at elevated temperatures, and the use of low energy oxygen ion bombardment to enhance the reoxidation process, as well as the collection of sufficient data to permit the use of luminescence as a characterization method for inaccessible components in future fusion devices and present fission reactor experiments. Two types of α-alumina have been studied; Deranox 995 and Wesgo 995. Surface electrical conductivity and He+ IBIL are presented before and after different thermal annealing cycles in air, oxygen, and during oxygen ion bombardment. In the initial stage of degradation, recovery is enhanced by increasing annealing temperature and time, as well as exposure to oxygen. However oxygen bombardment is only effective for severe degradation. One concludes that machine venting to air is a practical way to mitigate surface degradation in situ.

Id 637

Abstract Final Nr. P1.164

Structural changes induced in silica by ion irradiation observed by IR reflectance spectroscopy

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The structural changes produced by ion irradiation in different types of silica were observed by IR reflection spectroscopy. The change in the surface structure of the implanted face of a silica sample was studied monitoring the changes in the wavenumbers of fundamental structural bands. The position of the Si–O stretching band (at wavenumber ~1122 cm-1) is directly correlated with the average Si–O–Si bond angle in the silica structure. At wavenumbers lower than 2000 cm-1 (wavelengths greater than 5µm) due to strong absorption in this region, the observation of absorption of the Si–O stretching IR band requires very thin specimens. For bulk silica samples the observation reflection mode of asymmetric stretching band can be made since only a thin surface layer of the specimen is observed In the present work the structure modification caused by self-ions (O, Si) and He implantation at different fluences has been measured in three types of fused silica with different OH and impurity content (KU1, KS-4V and Infrasil 301). In these measurements the predominant process of energy transfer is the electronic excitation. KU1 and KS-4V are high purity synthetic silica, that are candidate materials for optical components in fusion devices due to their radiation hardness After ion implantation the IR reflection spectra of the implanted face was measured and the spectra were compared with the unimplanted one. The spectra of the three types of silica implanted with the same ion and fluence between 400 and 1400cm-1, is similar (no dependent on OH or impurity content of silica). The IR reflection spectra of neutron irradiated samples at fluences 1021n/m2 and 1022n/m2 were also measured and compared with spectra of implanted samples, minor changes were observed in the reflection spectra of neutron irradiated samples

Id 582

Abstract Final Nr. P1.165

Fusion Material Irradiation Experiments under Magnetic Field Conditions

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To date, studies of structural and functional materials for fusion have focused on material behaviour as a function of irradiation dose, energy, temperature, etc. However, the performance of materials under such conditions when a magnetic field is present is still unknown. In magnetic confinement fusion reactor devices the materials will support intensive and hazard radiation environments under intensive magnetic field. In principle, the micro-structural and mechanical properties of materials are modified by the propagation of the defects produced by irradiation, and it is hypothesized that such propagation may be affected by strong magnetic fields. Therefore, improved experimental knowledge in structural materials is sought concerning the mobility, recombination, clustering or dissociation of defects, in order to increase knowledge in this area. The work presented here has been undertaken at the Centre for Micro Analysis of Materials whose main experimental tool is an electrostatic ion accelerator, devoted to the analysis and modification of materials. A custom sample holder, with an embedded permanent magnet, has been designed and built for performing irradiation experiments in an ambient magnetic field of 0.8 T over a broad range of temperatures (from -100 ºC to 200 ºC, using a LN cooled cold finger and an internal heater). The experiments consist of irradiating samples (Fe-Crx model alloys) with different Fe ions, from 1-10 MeV and 10-100 nA, to induce fusion relevant damage (up to 5 displacements per atom, dpa). In order to characterize the net effect of a magnetic field on defects evolution, specimens of each alloy are irradiated with and without the presence of the magnetic field. For this, the sample thickness, irradiation times and conditions are calculated so that each sample accumulates sufficient damage throughout the desired depth. Finally, analysis of the irradiated specimens is performed by Mössbauer and Positron Annihilation Spectroscopy techniques.

Id 860

Abstract Final Nr. P1.166

Short-range order effects on ion irradiated Fe-Cr and its impact on resistivity properties of RAFM alloys.

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Chromium is one of the fundamental components of reduced activation ferritic/martensitic (RAFM) steels. Which are intended to be used as structural materials in future fusion reactors. It provides to the steel their properties of protection against corrosion and good mechanical behavior under irradiation, leading to a minimization of embrittlement. EUROFER steel has an optimized concentration of Cr around 9%. This choice of the steel composition becomes a compromise between the two mentioned desired effects. Understanding the physical mechanisms of radiation responsible of property changes effectively help to develop radiation resistant materials. The hardening (and therefore embrittlement) is a result of loops created by long term aggregation of point defects. Therefore a complete understanding of point defects and chromium evolution is needed. We are observing that proton irradiation at low temperatures (over 300 K) enhances Cr re-arrangement in the lattice, even at quite low doses. This phenomenon is also called short-range order effect (SRO). Residual resistivity of Fe-Crx (x = 0.05, 0.1, 0.14) has been measured before and after 400K low dose proton irradiation. The observed changes in resistivity confirm that at proportions over 10%, Cr tends to aggregate and at lower proportions it tends to spread within the lattice. In this work we will also present resistivity recovery (RR) curves of Fe-10%Cr samples with different initial SRO parameter and how the resistivity recovery depends on this Cr ordering. As this technique has been used as a tool to interpret the dynamics of primary defects created by radiation, this work stresses the importance of the Cr initial state when trying to validate computational simulations and its implication for validation of radiation damage in the context of the multi-scale approach.

Id 962

Abstract Final Nr. P1.167

Recrystallization kinetics of pure tungsten warm-rolled to different strains during annealing in the temperature range 1150 °C to 1350 °C

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Pure tungsten is considered as potential candidate material for the plasma facing first wall and the divertor of future fusion reactors. Both parts have to withstand high temperatures during service which will alter the microstructure of the material by recovery and recrystallization and cause degradation of the excellent material properties as a loss in mechanical strength and embrittlement. The thermal stability of two pure tungsten plates warm-rolled to different strains (67% and 90% thickness reduction respectively) was investigated by isothermal annealing in the temperature range between 1150 °C and 1350 °C. During annealing, recovery and recrystallization processes occur; the accompanying changes in mechanical properties are quantified by Vickers hardness measurements. After an initial recovery stage with only a slight hardness decrease, the hardness drastically drops following the characteristic sigmoidal shape for recrystallization. The microstructural changes during annealing are characterized by electron backscattered diffraction: the initially weak texture of the warm rolled plate is replaced by a random recrystallization texture. The recrystallized volume fraction is determined quantitatively and the recrystallization kinetics analyzed in terms of Johnson Mehl Avrami Kolmogorov kinetics. A much lower activation energy for recrystallization is obtained for the highly-strained plate, suggesting that warm-rolling of tungsten to larger rolling reductions leads to faster degradation of its mechanical properties during service. The observed time spans for recrystallization and the obtained values for the activation energy of the recrystallization process, nevertheless, indicate a sufficient thermal stability of both tungsten plates during operation at 800 °C.

Id 465

Abstract Final Nr. P1.168

Small-angle neutron scattering (SANS) characterization of 13.5 Cr oxide dispersion strengthened ferritic/martensitic steel for fusion applications

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This contribution presents the results of a microstructural characterization, by means of small-angle neutron scattering (SANS), of laboratory heats of a 13.5 Cr wt % ferritic/martensitic oxide dispersion strengthened (ODS) steel with variable Ti and W contents, introduced as alloying elements to improve the mechanical performance of these materials in view of application for fusion technology. The different heats were produced by mechanical alloying and subsequent heat treatment (2 h 1050°C), with fixed Y2O3 content 0.3 wt%, W contents of 1 and 2 wt% and Ti contents varying between 0 and 0.4 wt%. Samples for SANS investigation were prepared as platelets approximately 1 cm2 in surface area and 1 mm thick. The SANS measurements were carried out at the D22 instrument at the High Flux Reactor of the Institut Max von Laue – Paul Langevin, in Grenoble; an external magnetic field of 1 T was applied to the samples in order to saturate their magnetization and measure the nuclear and the magnetic SANS components. The selected experimental conditions allowed the measurement of the SANS cross-sections over an angular range corresponding to particles sizes ranging approximately between 10 Å and 500 Å. The results show that increasing the Ti content a progressive increase in the SANS cross-section is observed in an angular range corresponding to particles sizes between 10 Å and 50 Å approximately, which is attributed to the developement of Ti rich nano-clusters such as nuclei of Y-Ti-O phases. At lower angles, corresponding to much larger particle sizes, the effect on the SANS cross-section is more complex, revealing anyhow a decrease of particle size and volume fraction for maximum Ti content. This is attributed to the modification of large Cr carbides. All these results will be discussed also with reference to other metallurgical information obtained by electron microscopy on these same materials.

Id 274

Abstract Final Nr. P1.169

Pre-analysis of the Copper Neutronics Benchmark experiment for nuclear data validation

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Copper is an important heat sink material for fusion power reactors and it is also used for diagnostics, microwave waveguides and mirrors in ITER. Fusion relevant experiments on copper are lacking: there is only an OKTAVIAN experiment on a Cu spherical shell with measurement of the leakage spectrum. A new benchmark experiment on pure copper assembly is presently underway at the Frascati Neutron Generator (FNG) aimed at testing and validating the recent JEFF and FENDL neutron cross section data for fusion applications under 14 MeV neutrons irradiation. In the proposed experiment, relevant neutronics quantities (e.g. reaction rates, neutron flux spectra, nuclear heating, etc.) will be measured using different experimental techniques to get the nuclear quantities of interest (e.g. activation of selected set of foils, neutron and gamma spectrometry, unfolding, dosimeters, etc.). The Copper block has been designed and assembled on the basis of the pre-analysis presented in this work, performed with the MCNP5 Monte Carlo code, using JEFF 3.1.1. nuclear data library for transport and IRDF2002 for activation foils. The pre-analysis has been performed in order to define suitable dimensions of the copper block and optimise the experimental set-up for the measurements. This includes the irradiation conditions, which are required to obtain measurable quantities with the activation foils techniques, as well as the definition of the detectors positions. Furthermore the design has been optimised to reduce the effect of the background due to the neutrons back-scattered from the wall. The insertion of a polyethylene shield has been considered for background reduction purpose. The results of this pre-analysis show that a copper block with dimensions of 60x70x60 cm3 results to be the best compromise between performance and costs. The use of the polyethylene shield is ineffective because it perturbs the neutron spectrum more than the background itself does.

Id 116

Abstract Final Nr. P1.170

Rami assessment for IFMIF lithium facility

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A summary of a RAMI (Reliability, Availability, Maintainability and Inspectability) assessment performed on the IFMIF (International Fusion Materials Irradiation Facility) Lithium Facility (LF) is presented. Given the high availability requirements for the IFMIF plant, a target requirement of 94% availability has been attributed to the LF facility and its feasibility verified for the proposed system design. The facility reliability and availability performance for the foreseen operation mission has been investigated by the mean of reliability block diagrams. The performed RBD analysis shows that the LF could be able to comply with such request, as the obtained mean availability for ten years of simulation is of 93.35%. The Reliability to operate the LF without failure during the yearly scheduled operation time is about 64%. Most critical components in terms of impact on LF reliability and availability, are: i) the Li loop Target Section (back plate) and Electro-magnetic pumps and, ii) the ICS Common loop Getter Beds of the Hydrogen HT and EMP for the impurity control function. The Impurity Control System (ICS) of LF is available for about 98% of the year to operate its control function on the Li impurity levels for a correct performance of the Target. Particularly, the ICS is able to perform the detritiation of the liquid Lithium during the LF operating phase with a Reliability of about 80%. The impact of alternative design solutions on system reliability has also been evaluated by increasing/reducing redundancy for some key components to enhance reliability or reduce system costs with no performance loss. In particular, the use of two electro-magnetic pumps (EMPs) in the Li loop and the use of redundant Nitrogen Traps and Cold Traps to operate during the LF shutdown phase have been investigated.

Id 838

Abstract Final Nr. P1.172

Analysis of gaps in the steels database for EUROFER as structural material for DEMO

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Steels in DEMO machine will act as structural material, requiring high performance. Doses up to 20 dpa for the First Wall of an 'Early DEMO' are expected, followed by a second phase of where total dose will reach up to 50 dpa. Steels will have to operate at different temperature ranges depending of the coolant selection: an operation range between 250 - 350°C is associated to water Breeder Blanket solutions whereas temperatures between 350°C and 550 are foreseen for Helium coolant options. In addition, DEMO will be a pulsed machine, with pulses longer than 2 hours. Determining performance of material during off-normal conditions and the effect of off-normal loads on subsequent material behavior is also required. EUROFER steel was developed in Europe to operate in optimum conditions for Helium cooled options. Characterization of EUROFER97 has been done with the purpose of its qualification to be used in the TBM in ITER up to few dpa. Other experiments at higher doses have been done reaching 80 dpa, but not at all temperatures of interest. This paper analyses the gap for full qualification of the Eurofer steel in particular in the region between 300°C and 400°C where the ductile to brittle transition takes place. Furthermore, interaction between the material community, component designers and structural design criteria developers is paramount; only in this way can a cohesive work programme be produced to deliver designs able to meet system requirements, with validated performance and qualified materials. Besides conventional EUROFER steel, ODS Feritic and martenstic steels have also been developed in Europe at semi–industrial scale. The main use of this kind of steels would be applications at higher temperature, up to 750°C. Their properties have been partially evaluated, taking EUROFER as reference Material, and further lines of action have been identified.

Id 636

Abstract Final Nr. P1.173

HIP-Welding tests for the molybdenum 1st mirror of CXRS-Spectroscopy

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One major part of the CXRS-Spectroscopy is the first mirror M1. The first mirror is responsible for acquisition and transportation of the optical signal in the required quality and it is therefore placed near the plasma. The mirror is working in the severe radiation and particle environment and should uphold its optical properties over a long period. Erosion and deposition are the main effects which are influencing the optical properties of the first mirror. Single crystal molybdenum was chosen as the preferable mirror material, because of its stability against erosion and its resistance for sputtering. The size of the planned mirror is 126 x 280 x 20 mm. At present, it is technologically not possible to produce samples of single crystal molybdenum in the required high quality with a diameter larger than 150 mm. As a result, the planned mirror shall be composed of three plates of single crystal molybdenum connected by means of HIP welding. The HIP-process is based on the effect of high temperature and high pressure on metal at the same time. The welding process occurs just below the solidus temperature and at a high pressure. This paper describes the tests to determine the necessary welding parameters and type of connection for the joining of Mo/Mo and of Mo/TZM. In the first case, three disks with a diameter of 25 mm and t = 6 - 8 mm are stacked in the order of Mo-Mo-TZM and should be joined by HIP. In the second case the disks are arranged in the same way, but there is an intermediate layer of powdered molybdenum between the disks. First tests for polycrystalline molybdenum are carried out at a temperature of 1000°C, a pressure of 1000 bar and an exposure time of 4 hours. The upper temperature limit is 1200°C.

Id 615

Abstract Final Nr. P1.174

Creep irradiation testing of copper alloy for the ITER First Wall Panels

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One of the major challenges to establish technologies for the fusion energy is the development of appropriate in-vessel materials to be used in ITER and later on in DEMO. The materials shall resist neutron fluence, heat flux, thermo-mechanical and electro-magnetic loads at an elevated temperature regime for a specified lifetime. While promising options have been selected through extensive past material research within the scientific fusion community, the aspects of manufacturing and quality assurance at large scale are typical contributions by industrial partners. Usually material characterisation on industrial scale is based on standard sized test specimens. But limitations because of the available material to be tested and targeted irradiation facility may require sub-size specimens in the range of 27x5 mm or below. Therefore, the test procedures have to be established according to respective boundary conditions set by the available material irradiation facilities. Cooperation between the research centre SCK•CEN Mol and the TÜV Rheinland Industry Service GmbH has been launched by Fusion for Energy (F4E) and ITER International Organization (IO). The work is focused on the material testing of irradiated and reference specimens of base metal CuCrZr and its HIPed joints with 316L(N)-IG. From this combined scientific and industrial point of view the efforts including a novel design of an in-pile pre-stressed specimen, corrosion inhibition, reactor irradiation and first results for material tests on irradiated CuCrZr material and the corresponding reference tests are presented. The test campaign consists of: (i) in-pile creep specimens, where a test specimen is axially loaded and exposed for neutron irradiation up to 0.7 dpa and its plastic deformation due to thermal creep and irradiation is measured and (ii) creep tests after irradiation up to 0.7 dpa aiming at 7 days of creep up to rupture. Both tests are compared with reference tests with non-irradiated specimens. The material specimens are made of CuCrZr and its HIP joints with 316L(N).

Id 449

Abstract Final Nr. P1.176

The design of the Conventional Facilities of IFMIF

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IFMIF, the International Fusion Material Irradiation Facility, presently in its Engineering Validation and Engineering Design Activities (EVEDA) phase, framed by the Broader Approach Agreement between Japan and EURATOM, has accomplished in 2013, on schedule, its mandate on the engineering design of the plant. 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 EVEDA phase. The successful operation of such a challenging plant, demands careful assessment of the Conventional Facilities (CF), that holding adequate redundancies will allow the target plant availability. The present paper addresses the design proposed in the IFMIF Intermediate Engineering Design Report regarding the CF with particular attention to IFMIF's outdoor layout configuration, the Electrical Power Supply, the Heating Ventilation and Air Conditioning, the Heat Rejection System, the Service Water System, the Service Gas System (Ar, He, compressed Air& Nitrogen), the Exhaust Gas Processing System, the Radioactive Waste Treatment, the Fire Protection System and the Central Control & Common Instrumentation. This design was developed progressively; firstly by establishing a conceptual design starting from system functional description, followed by the identification of the corresponding interfacing systems and their technical requirements, and ending with the definition of the process flow diagram and basic equipment layout. Once the technical requirements were identified and the design basics established the system design were further developed. Piping & instrumentation diagram (P&ID's) and equipment list for different systems, as well as a layout plan of the equipment and routing of the air ducts, piping and cable trays were defined and eventually integrated into the 3D model of the Building.

Id 483

Abstract Final Nr. P1.177

RAMI analysis program design and research for CFETR (Chinese Fusion Engineering Testing Reactor) Tokamak Machine

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CFETR is a test reactor which shall be constructed by National Integration Design Group for Magnetic Confinement Fusion Reactor of China with an ambitious scientific and technological goal. The Reactor has the equivalent scale compared with ITER, but has the complementary function to ITER. CFETR is a demonstration of long pulse or steady-state operation with duty cycle time not less than 0.3~0.5 and the full cycle of tritium self-sustained with TBR not less than 1.2. At the same time it will be exploring options for DEMO blanket and divertor with an easy changeable core by remote handling way. To be able to reach its scientific and technological objectives, as one of technical risks control methods, RAMI analysis need to be done during the hold lifetime of CFETR, from conception design to decommissioning. Base on stating of CFETR lifetime and preliminary operational programme, the RAMI analysis program and process are designed and discussed, it consists of five major steps: (1) Functional analysis are performed, (2) Calculating reliability block diagrams, (3) Analyzing Failure Mode, Effects and Criticality Analysis (FMECA), (4) Risk mitigation actions are taken to ensure every system is compatibility with RAMI objectives, (5) All the RAMI analysis are integrated as the final RAMI analysis reports to be reviewed in the system final design review. Along with the elements of the analysis the Vacuum Vessel System was performed to provid as examples, detailed showing how the CFETR RAMI analysis is carried out. CFETR RAMI analysis guidelines were designed and established, after constantly revised and improved these analysis criteria and programs will become the basis standards for CFETR RAMI analysis. Preliminary RAMI analysis of CFETR VV system was obtained, which will be updated with the VV system design progresses.

Id 182

Abstract Final Nr. P1.178

Shutdown Dose Rate Analysis During Maintenance Scenarios of the Neutral Beam Injectors

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Please find attached my abstract for a proposed paper for your consideration. Not only do I feel my paper would be of particular interest to the fusion neutronics community but I also feel it will be of interest to the wider fusion community. As designs for a viable fusion power plant progress limiting the maintenance and shutdown times will become of greater importance. With the additional flexibility added to the MCR2S code we are now able to accurately model a given maintenance scenario by moving multiple activated components to various locations. This will allow the shutdown dose rate during maintenance to be ascertain and help with the production of efficient maintenance procedures.

Id 133

Abstract Final Nr. P1.179

The Resilience of an Operating Point for a Fusion Power Plant

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The operating point for fusion power plant design concepts is usually determined by simultaneously satisfying the requirements of all of the main plant systems and finding an optimum solution, for instance the one with the lowest capital cost or cost of electricity. This is a static assessment which takes no account of the resilience of that operating point to variations in key parameters and therefore includes no information about how difficult to control the chosen operating point may be. Control of the operation point of a fusion power plant is a large subject with much work still to be done, and is expected to play an increasing role in the future in choosing the optimum design point. Here we present results of one analysis which relates to the ability to load-follow, that is to vary the power production in the light of varying demands for power from the electricity network, over the range 0.5-1 GWe. The operating point chosen as the optimum static point may in fact not be a good one to allow load following since changing the power output by varying the density, temperature or fuel mix can lead to large changes in other parameters such as the confinement time, plasma current, or current drive power. Here we look at the sensitivity of such an operating point to key assumptions, showing how the design point has a different optimum when the requirements of load-following are imposed. A method of selecting a new optimised operating point, which takes into account this resilience to changes, is proposed. This work was part-funded by the RCUK Energy Programme and by the European Union's Horizon 2020 programme

Id 529

Abstract Final Nr. P1.180

Development and application of a secondary surface source mesh routine in MCNP6 and its application to fusion-relevant radiological field mapping

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High-fidelity predictions of the radiation fields associated with fusion devices are needed to support safety requirements. Supporting radiation transport calculations for fusion problems can benefit from a 2-stage modelling approach; 1) the mapping of a field generated in a complex region or component onto bounding surfaces, and 2) using the mapped surfaces in a second-stage calculation, typically over a large distributed volume. For example one may predict the dose levels in detail inside a tokamak device, such as JET or ITER, then using the 'recorded' source in a secondary calculation the doses in the surrounding complex can be modelled. A routine has been developed which utilises the MCNP surface source write file to enable the generation of a discretised, portable bounding surface source which overcomes some of the difficulties encountered with the native MCNP surface source read/write capability. The method devised approximates the original source in space, energy and angle, as opposed to an exact reproduction of the particles. This methodology has been implemented as a FORTRAN subroutine, using the wssa file created by the native MCNP surface source write function. Source particles are generated on a mesh, based on probability distribution functions generated from the position, energy, weight, and direction vector values from the wssa file. Calculations using this new routine show that results are within 8% of the original MCNP calculations. The generated source is independent of the geometry and can be translated to any position, making it an ideal tool to use to predict doses during transfer of radioactive material. The new routine has been applied to fusion scenarios, such as the radiological field mapping of the planned JET DT campaign and for the transport of activated divertors, blankets and diagnostic port plugs in casks around the ITER complex. Calculations of the casks using this new routine have shown a peak dose of 44.93 Gy during their transportation to the hot cell. Improvements in the computational efficiency 'figure of merit' have been shown to be better than a factor of 10 for this new routine. This work was funded by the RCUK Energy Programme under grant EP/I501045.

Id 567

Abstract Final Nr. P1.181

The Conceptual design of WDS in CFETR

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During the CFETR operation, including maintenance campaigns, tritiated water will be generated by various sources, which lead to accumulated amounts that are by much in excess of the amounts that can be periodically discharged as effluent. The capacity of the WDS is based on processing the amounts of tritiated water generated during normal operation including maintenance. The required processing capacity for tritiated water is about 60 kg/h based on the various source terms in CFETR. The WDS of CFETR is composed of the following subsystems: •Front-end process •Catalytic isotope exchange process •Electrolysis process •Hydrogen stream process •Oxygen stream process •Utility subsystems •Tritiated water holding tank system The first to fifth subsystems recovers tritium from tritiated water and reduces the remaining tritium concentration and accumulated amounts in effluent streams to within the discharge limits. The tritiated water holding tank system stores the various tritiated water streams according to their tritium concentrations in a system of tanks located in the basement of the tritium building, prior to processing.

Id 370

Abstract Final Nr. P1.183

RAMI Analysis for DEMO HCPB blanket concept cooling system

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Reliability and availability targets for DEMO plant will play a key role in the engineering development of the reactor. Present work focuses on the RAMI (Reliability, Availability, Maintainability and Inspectability) assessment for the HCPB (Helium Cooled Pebble Bed) blanket cooling system as one of the foreseen conceptual options for DEMO plant. RAMI assessment was performed on the base of currently available design sheets for HCPB cooling system. The following sub-systems were considered in the analysis: blanket modules, primary cooling loop including pipework and steam generators lines, pressure control system (PCS), coolant purification system (CPS) and secondary cooling system. For PCS and CPS systems no HCPB specific design was available, so that an extrapolation from ITER Test Blanket Module corresponding systems was used as reference design in the analysis. Moving from plant components failure modes identified by the mean of a Failure Mode and Effect Analysis (FMEA) assessment, the HCPB system was modelled using reliability block diagrams (RBD) taking into account: system reliability-wise configuration, operating schedule currently foreseen for DEMO, maintenance schedule and plant evolution schedule (i.e. scheduled substitutions of blankets etc.) 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 30 calendar years. The resulting availability performance was finally compared to availability goals previously proposed for DEMO plant by a panel of experts. The study suggests that inherent availability goals proposed for DEMO PHTS system and Tokamak auxiliaries are potentially achievable for the primary loop of the HCPB concept cooling system, but not for the secondary loop. A sensitivity analysis is also presented to explore results dependency on key estimated parameters and analysis assumptions.

Id 385

Abstract Final Nr. P1.184

Revisiting the analysis of passive plasma shutdown during an ex-vessel loss of coolant accident in ITER blanket

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In this contribution, the analysis of passive safety during an ex-vessel loss of coolant accident (LOCA) in the first wall/shield blanket of ITER has been studied with AINA safety code. In the past, this case has been studied using robust safety arguments,[1,2,3] based on simple 0D models for plasma balance equations and 1D models for wall heat transfer. The conclusion was that, after first wall heating up due to the loss of all coolant, the beryllium evaporation in the wall surface would induce a growing impurity flux into core plasma that finally would end in a passive shut down of the discharge. The analysis of plasma-wall transients in this work is based in results from AINA code simulations. AINA (Analyses of IN vessel Accidents) code is a safety code developed at Fusion Energy Engineering Laboratory (FEEL) in Barcelona[4]. It uses a 0D-1D architecture, similar to that used for past analyses. For this study, some AINA models have been upgraded. The wall model has been changed to adapt it to current ITER blanket design and blanket modules distribution. A model for the view factors of different blanket modules, used in the calculation of synchrotron power reflection, has been implemented. An improved model for neutron power deposition in the blanket has been also implemented. The impurity transport model has also been studied. It is shown that a slight variation in the model hypotheses has a strong influence in the duration of the plasma transient and thus in the final wall temperature. The results show good agreement with previous studies, provided some hypothesis are assumed in the models. Plasma shut down happens before significant melting occur in first wall. However, there is a narrow temperature margin, and it cannot be assured that some areas of the blanket won't be working beyond the thermal operating limit. [1] Amano, T. "Passive shut down of ITER plasma by Be evaporation" National Institute for Fusion Science, Research Report, Vol.402, 1996 [2] Honda, T., et al., "Analyses of Passive Plasma Shutdown during Ex-Vessel Loss of Coolant Accident of First Wall/Shield Blanket in Fusion Reactor", J. Nucl. Sci. Technol.(1997), vol. 34, no6, pp. 538-543 [3] Bartels, H.W., "Shutdown temperature due to Be evaporation from FW for ITER FEAT", ITER Garching JWS, G81 MD 4, (March 2000) [4] J. Dies, et al., "AINA safety code, v3.0", FEEL-UPC Internal Report, June 2013, 85p.

Id 744

Abstract Final Nr. P1.185

Status of the EU Test Blanket Systems Safety Studies

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The European Joint Undertaking for ITER and the Development of Fusion Energy ('Fusion for Energy'- F4E) provides the European contributions to the ITER international fusion energy research project. Among others it includes also the development, design, technological demonstration and implementation of the European Test Blanket Systems (TBS) in ITER. An overview of the ITER TBS program has been presented recently at ISFNT-10. Currently two EU TBS designs are in the phase of conceptual design - Helium-Cooled Lithium-Lead (HCLL) and Helium-Cooled Pebble-Bed (HCPB). Safety demonstration is an important part of the work devoted to the achievement of the next key project milestone the Conceptual Design Review. The paper reveals the details of the work on EU TBS safety performed in the last couple of years in the fields of update of the TBS safety demonstration files; TBS Safety approach, design principles, requirements, features and safety functions; detailed TBS components classifications; Radiation Shielding and Protection; Selection and definition of reference accidents scenarios, and Accidents analyses; and Management and categorization of TBS radioactive waste. Finally the authors share the on-going and planned future EU TBS safety activities.

Id 292

Abstract Final Nr. P1.186

Concept Design Analyses of an ITER Radwaste System Pipeline

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As a part of ITER research and development activities, concept design of a pipeline containing low-level liquid wastage is being carried out. Since failure of the radwaste pipeline can result in release of radioactive materials and threat the safety of penetrating Tokamak and annex buildings, the pipeline should satisfy requirements of faulted condition as well as normal operating condition. The objective of this study is to examine load-carrying capacity of the radwaste pipeline through systematic numerical analyses. In this context, at first, static finite element analyses were performed for five candidate layouts under conservative seismic displacements. Subsequently, alternative dynamic analyses were conducted for the same layouts to confirm whether the simplified approach is valid or not. Based on the static analysis results such as maximum stresses and displacements, a spiral-type configuration was derived as the appropriate layout for detailed design of the radwaste pipeline. Meanwhile, both modal and response spectrum analyses results showed reasonable dynamic characteristics of the pipeline. Also, it was proven that the conservative static analysis method is comparable with the detailed dynamic analysis method from the conservative viewpoint.

Id 489

Abstract Final Nr. P1.187

Overvoltage Protection for Magnetic Systems during Disruption in Tokamak

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This paper presents an overvoltage protection system for magnetic systems in a tokamak. During a plasma disruption the magnetic flux in the tokamak changes rapidly, which is likely to cause high-voltage surges among the magnetic systems. Sometimes an overvoltage surge happens and the overvoltage wave can flow through the entire magnetic system device, which may even bring severe damage to the components if the overvoltage protection is not available. To deal with this problem, the overvoltage model of plasma disruption (OMPD) is introduced by analyzing the relationship between voltage and current with the specialized disruption model. The OMPD is proved very effective in further studies on plasma disruption and mitigation. The OMPD is applied to the Ohmic Heating System of J-TEXT, and derives some important parameters for overvoltage protection, like voltage-time characteristics and surge energy. An overvoltage protection system is implemented according to the analysis above. The protection system benefits from metal oxide voltage dependent resistor and intelligent switching control, and its design is validated with disruption experiments. Now the overvoltage protection system has been deployed in J-TEXT and serves well in daily experiment. Keywords—overvoltage protection; tokamak; plasma disruption; OMPD

Id 587

Abstract Final Nr. P1.188

Update of the MELCOR calculations for the validation of the ITER Cryostat design

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MELCOR is a fully integrated, engineering-level computer code whose primary purpose is to model the progression of accidents in nuclear power plants. This paper presents a series of updates to the MELCOR calculations for three sets of events that affect the ITER Cryostat: Cryostat Ingress of Coolant (He), Cryostat Loss of Vacuum and Helium leak in the Galleries. The objective of the new calculations is to generate a set of thermal and pressure inputs appropriate for the validation of the mechanical design of the Cryostat. For that purpose, a new MELCOR nodalization based on the Finite Element Model (FEM) of the Cryostat has been created. The geometry breakdown has been done using controlled configuration models (ENOVIA) in order to identify geometry changes and to account for up-to-date free volumes, surfaces, etc. In addition, the effect of conduction between MELCOR heat structures has been evaluated. With these changes implemented, the Helium leak and cryostat air ingress events have been simulated. The results are the time histories of the temperatures and pressures in the different cryostat parts and volumes. The heat transfer coefficients at the cryostat surfaces have been also calculated as a function of time. These results will be used as input data for the FEM analysis of the cryostat.

Id 405

Abstract Final Nr. P2.001

Manufacturing, assembly and tests of SPIDER Vacuum Vessel to develop and test a prototype of ITER Neutral Beam Ion Source

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The SPIDER experiment (Source for the Production of Ions of Deuterium Extracted from an RF plasma) aims to qualify and optimize the full size prototype of the negative ion source foreseen for MITICA (full size ITER injector prototype) and the ITER Heating and Current Drive Injectors to be installed at ITER site. Both SPIDER and MITICA experiments are presently under construction at Consorzio RFX in Padova, with financial support from IO (ITER Organization), Fusion for Energy, italian research institutions and contributions from Japan and India Domestic Agencies. The vacuum vessel hosting the SPIDER beam source has been manufactured, tested and assembled on site during the last two years 2013-2014. The cylindrical vessel, about 6 m long and 4 m in diameter, is composed of two cylindrical modules and two torispherical lids at the ends. All the parts are made by AISI 304 stainless steel. The possibility of opening/closing the vessel for monitoring, maintenance or modifications of internal components is guaranteed by bolted junctions and suitable movable support structures running on rails fixed to the building floor. A large number of ports, about one hundred, are present on the vessel shell for diagnostic and service purposes. The main working steps for construction and specific technological issues encountered and solved for production are presented in the paper. Assembly sequences and tests on site (vacuum and functional tests) are furthermore described in detail, highlighting all the criteria and requirements for correct positioning and testing performances.

Id 392

Abstract Final Nr. P2.002

The Earthing System of the PRIMA Neutral Beam Test Facility based on the Mesh Common Bonding Network Topology

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PRIMA is a large experimental facility under realization in Padova, aimed to develop and test the Neutral Beam Injectors for the ITER experiment. Two separate experiments, SPIDER and MITICA, will be hosted in the PRIMA facility. MITICA will be the first full size ITER injector aiming at operating up to the full acceleration voltage (1 MV) and at full power (16.5 MW). The operation of these devices involves high RF power and very high voltage. Frequent and intense electrical breakdowns inside the beam sources occur regularly. Several sensitive diagnostics systems are needed for control and characterization of the physical phenomena, which must operate in this hostile environment. The presence of a widespread and carefully optimized earthing system is of paramount importance to achieve a satisfying immunity from disturbances for equipment and diagnostics. The paper describes the design and the realisation of the earthing system of the PRIMA facility, which is based on the MESH-Common Bonding Network (MESH-CBN) topology, as recommended by IEC and IEEE standards (IEC 61000-5-2, IEC 62305, IEEE 1100, IEE 142) for installations where high levels of Electromagnetic Interferences are expected. A similar approach was also chosen for ITER, so that PRIMA will represent a useful test bed for these techniques in conditions very similar to the ITER ones. The principles of the MESH-CBN approach were adapted to the PRIMA layout, which is composed by several buildings, that are independent for seismic and architectural reasons, but are linked by many electrical conduits and hydraulic pipelines. The presence of two independent experiments was taken into account, as well as the availability of huge foundations with a large number of piles and pillars. Buildings parts dedicated to host control rooms and sensitive equipment were treated with particular care.

Id 817

Abstract Final Nr. P2.003

The SPIDER Cooling Plant from design to realization

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This paper presents a description of the design and construction of Cooling Plant (CP) for SPIDER (Source for Production of Ion of Deuterium Extracted from RF Plasma) experiment, under realization at the ITER Neutral Beam Test Facility (NBTF) in Padua, Italy. This supply is presently under procurement with Fusion for Energy Contract F4E-OPE-351 – "Cooling Plant for MITICA and SPIDER experiments". Starting from the requirements specified in the Technical Specifications of Cooling Plant Procurement, the supplier Delta-Ti S.p.A. has developed a detailed and final design consisting of three different parts: hydraulic and general layout, electrical and control, chemical control systems. The main scope of the Cooling Plant is to reject the heat from in-vessel components taking into account also other technical requirements as the use of ultrapure deionized water as insulating coolant, the accurate thermal control of some in-vessel components, calorimetric measurements for scientific purposes. This paper presents the main characteristics of the SPIDER Cooling Plant inside PRIMA buildings, the connection of the Cooling Plant to the components inside the SPIDER vacuum vessel, the optimization process of the two large water basins by means of CFD simulations and, finally, details about present procurement status and foreseen tests.

Id 297 (v2)

Abstract Final Nr. P2.004

Design and radiation protection aspects for the PRIMA Cooling Plant

Francesco Fellin (a), Pierluigi Zaccaria (a), Manuela Battistella (a), Samuele Dal Bello (a), Marco D'Arienzo (b), Sandro Sandri (c), Angela Coniglio (d), Gilbert Agarici (e), Giovanni Dell'Orco (f), Roberto Bozzi (g), Gabriele Cenedella (g)

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This work presents a description of the radiation protection issues impacting on Primary Cooling circuits of the MITICA (Megavolt ITER Injector Concept Advanced) experiment in Padova. MITICA represents the full scale Neutral Beam Injector of ITER; two Heating neutral Beams are planned to be used in Cadarache (F) to reach the plasma nuclear fusion conditions, in addition to other heating systems. Three actively cooled components will be installed in the MITICA Beam Line Vessel, and their surfaces will be impacted by high energy neutral particles produced by an RF Ion Source and accelerated up to 1 MV by a set of five acceleration grids. Ultrapure and deionized water will be used as an insulating coolant to guarantee high voltage holding. Due to the high velocity inside water channels and due to the components materials, the production of activated corrosion products is likely to pose safety concerns in MITICA facility. Considerations about this issue and technical solutions to be adopted for the Cooling Plant are described in this paper.

Id 297 (v2)

Abstract Final Nr. P2.005

Manufacturing of the full size prototype of the ion source for the ITER neutral beam injector. The spider beam source

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Each of the two ITER Heating Neutral Beam injectors (HNBs) will deliver 16.5MW of heating power obtained by the acceleration of a 40A deuterium negative ion beam up to the energy of 1MeV. The ions are generated into a cesiated ion source, where the gas is ionised with a radiofrequency electromagnetic field. The largest world experiments for the production of negative ion beam presently operating in IPP Garching is ELISE, which represents the first and bigger step from the small size IPP devices to half of the size of HNB ion source. The technology developed in IPP has been applied to the development of another big experiment, SPIDER, which will be the “ion source” test facility for the full size ITER neutral beam ion source. The SPIDER beam source comprises an ion source with 8 radio-frequency drivers and a three grid system, providing an overall acceleration up to energies above 100keV. The beam source will be housed in a vacuum vessel which will be equipped with a beam dump and a graphite calorimeter to diagnose the beam. The manufacturing design of the main parts of the SPIDER Beam source has been completed and also the tests on the prototypes have been successfully passed. The most delicate and complex parts of the ion source and the accelerator, developed by galvanic deposition of copper are being manufactured. The parts will be completed by the end of 2014, when the assembling of the challenging device, will start. The paper describes the adaptations operated on the design of the beam source for the fabrication, in particular also to the engineering development to enable the fulfilment of the tight requirements set in the technical specifications. Moreover the tests performed on the prototypes and the modifications that were implemented are reported.

Id 861

Abstract Final Nr. P2.006

Plant integration of MITICA and SPIDER experiments with auxiliary plants and buildings on PRIMA site

Francesco Fellin, Marco Boldrin, Pierluigi Zaccaria, Piero Agostinetti, Manuela Battistella, Marco Bigi, Mauro Dalla Palma, Samuele Dal Bello, Aldo Fiorentin, Adriano Luchetta, Alberto Maistrello, Diego Marcuzzi, Edoardo Ocello, Roberto Pasqualotto, Mauro Pavei, Andrea Rizzolo, Vanni Toigo, Matteo Valente, Loris Zanotto, Luca Calore, Federico Caon, Massimo Caon, Michele Fincato, Gabriele Lazzaro, Michele Visentin, Enrico Zampiva, Simone Zucchetti

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This paper presents a description of the PRIMA (Padova Research on ITER Megavolt Accelerator) Plant Integration work, aimed at the construction of PRIMA Buildings, which will host two nuclear fusion test facilities named SPIDER and MITICA, finalized to test and optimize the Neutral Beam Injectors for ITER experiment. These activities are very complex: inputs coming from the experiments design are changing time to time, while the buildings construction shall fulfill precise time schedule and budget. Moreover the decision process is often very long due to the high number of stakeholders (RFX, IO, third parties, suppliers, domestic agencies from different countries). The huge effort includes: forecasting what will be necessary for the integration of many experimental plants; collecting requirements and translating into inputs; interfaces management; coordination meetings with hundreds of people with various and different competences in construction and operation of fusion facilities, thermomechanics, electrical and control, buildings design and construction (civil plants plus architectural and structural aspects), safety, maintenance and management. The paper describes these activities and also the tools created to check and to validate the building design, to manage the interfaces and the organization put in place to achieve the required targets.

Id 297 (v2)

Abstract Final Nr. P2.008

Design Optimization and Performances of New Sorgentina Fusion Source (NSFS) Supporting Materials Research

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In the framework of fusion materials research, a neutron source has been considered a key installation to support EU plan towards DEMO reactor design. IFMIF facility being the present proposal, a pragmatic approach to EU fusion roadmap timeline considers complementary solutions mandatory, within a shared strategy. New Sorgentina Fusion Source (NSFS) has been recently proposed in order to populate an engineering database through material irradiation tests. It takes advantage from proven technology of D-T neutron generators together with ion source and accelerator devices currently implemented in neutral injection systems at large experimental tokamaks. Deuterium and tritium enriched hydride is on-line reloaded by impinging D-T beams via ion implantation while high-speed rotating target improves heat removal and pulsed temperature allows D-T retention. Moreover, hydride metal layer is re-deposited increasing plant availability factor. In this contribution, a scaled target design is proposed to cope with thermal transients and mechanical loads. Solutions to significant thermal fatigue concerns are presented as well. Irradiation capability is then enhanced attaining relevant materials exposure. Main facility characteristics are provided as well as thermal and mechanical issues.

Id 720

Abstract Final Nr. P2.009

Steady state thermomechanical analysis of the IFMIF lithium target system with bayonet backplate under design conditions

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In the framework of European R&D fusion activities, a high flux neutron source is considered an essential device for testing candidate materials under irradiation conditions typical of future fusion power reactors. To this purpose, IFMIF (International Fusion Materials Irradiation Facility) project represents an important option to provide the fusion community with a source capable of irradiating materials samples at a rate of up to 20 dpa/fpy in a volume of 0.5 l. This is achieved by bombarding a high-speed liquid lithium target with a 10 MW double deuteron beam which yields a 14 MeV-peaked neutron spectrum. Within the engineering design work of the IFMIF/EVEDA project, which was concluded in half 2013, ENEA was in charge of the design of the lithium target system based on the so called bayonet backplate concept, which foresees the possibility to periodically replace only the most irradiated and thus critical component (i.e., the backplate) while continuing to operate the rest of the target for a longer period. With the objective of evaluating the structural performances of the system, an uncoupled thermomechanical analysis has been performed in collaboration with the University of Palermo by means of a qualified finite element code. An extensive neutronic analysis was also carried out through the MCNP transport code to determine the neutron and gamma heating which was then used as input in the thermomechanical model. In this paper, following a recent contribution [1] reporting calculation data obtained for nominal operating conditions, the results of the steady state thermomechanical analysis carried out under design conditions are presented. The outcomes have confirmed the general capability of the system to withstand the applied loads, although some potentially critical points are evidenced, which might require a further improvement of the system design before achieving an optimized configuration.

[1] P. Arena, D. Bernardi, G. Bongiovì, P.A. Di Maio, M. Frisoni, G. Miccichè, M. Serra, Engineering design and steady state thermomechanical analysis of the IFMIF European lithium target system, Proc. of SOFE-25 Conference, San Francisco, USA (2013) DOI: 10.1109/SOFE.2013.6635384

Id 821

Abstract Final Nr. P2.010

RFX radiological analysis and testing for improved tokamak and deuterium operation mode

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The RFX-mod device is a nuclear fusion facility based on the magnetic confinement of a plasma in the Reversed Field Pinch configuration, characterized by comparable toroidal and poloidal magnetic fields, with the toroidal field reversing near the edge. RFX-mod is equipped with a state-of-art active feedback control system of magnetic instabilities based don 192 saddle coils independently driven. Recently, such control system has been exploited to operate the device also as a Tokamak. In addition, the use of deuterium as feeding gas has been proposed. Such new operation regimes are likely to produce X radiation and neutrons. The licensing of the facility for the new operative configurations has been obtained at the end of 2012. At the completion of the licensing process a measurement campaing was performed to assess the good function of the radiological security systems. The two operation modes (tokamak and deuterium) were studied from the radiation protection point of view and both experimentally tested in two separated campaings. Special shielding requirements were evaluated for passages and ducts inside the controlled area. In both the experimental sessions, monitoring networks of passive dosimetric systems were placed in the areas designed to be accessible by personnel during the normal operation. Active monitors were used in parallel to control the spots more likely occupied by the staff. The testing process for the tokamak operation was performed with X and gamma ray monitors and dosimeters while for the deuterium operation also neutrons monitors and dosimeters were used. The short pulse of RFX-mod required specific measurement techniques, mainly for the counting integration with the active monitors. The results were completely acceptable for both the operation modes. The design assessment was confirmed with a security gap that is important for the operation to be initiated without any additional recommendation.

Id 933

Abstract Final Nr. P2.011

Study of the response of a piezoceramic motor irradiated in a fast reactor up to a neutron fluence of 2.77E1017 n/cm2.

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A piezoceramic motor has been identified as the potential apparatus for carrying out the rotation of the scanning head of a laser radar system used for viewing the first wall of the ITER vessel. This diagnostic is more simply referred to as IVVS (In Vessel Viewing System). The choice fell on a piezoceramic motor due to the presence of strong magnetic fields (up 8 T) and of the high vacuum and temperature conditions. To be compliant with all the ITER environmental condition it was necessary to qualify the piezo motor under gamma and neutron irradiation. In this paper are described the procedures and tests that have been performed to verify the compatibility of the operation of the motor adopted in the presence of a fast neutron fluence which was gradually increased over time in order to reach a total value of 2.77E17 n/cm2 . Such neutron fluence was obtained by irradiating the motor in a position close to the core of the fast nuclear reactor TAPIRO, in operation at the ENEA Casaccia Research Centre. The neutron spectrum in this position has been identified as representative of that found in the rest position of the IVVS head during ITER operation. The cumulative neutron fluence reached corresponds to that is expected to be reached during the entire life of ITER for the IVVS in the rest position without any shield. This work describes the experimental results of this test, the methodology adopted for the verification of the correct operation of the motor, and the methodology adopted to determine the total neutron fluence achieved.

Id 71

Abstract Final Nr. P2.012

Physical Program and Conceptual Design of the Diagnostics of the T-15 Upgrade Tokamak

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The Institue of Tokamak Physics of the Kurchatov Institute is upgrading now the T-15 tokamak to the machine with following parameters: R=1.5 m, a=0.67 m, B=2 T, Ipl= 2 MA and plasmas with elongation k95 2 and triangularity d95 0.4. The device will consist of the stainless steel single skin vacuum vessel, a carbon inner wall, will contain an ITER-like closed divertor and open upper divertor both cryopumped and covered by water-cooled carbon tiles. The magnetic system will consist of 16 toroidal and 6 poloidal water-cooled copper coils, capable of realizing Lower and Upper Single Null and Double Null magnetic configurations. The heating and current drive system consists of Neutral Beam Injection (3 co-injectors of each 2 MW/ 75 kV H0), ECRH/CD (7 Gyrotrons with each 1.5 MW at 2w in X-Mode with a possibility of EBW Heating and CD), ICRH/CD (3 antennas with each 2 MW including the possibility of the helicon wave generation), LHH/CD (2 Grills of 2 MW each) and is aiming at providing effective heating of both electrons and ions, and on- and off-axis CD. This new device with its favorable combination of low aspect ratio A=2.2 and reasonably high magnetic field B is beneficial for the plasma performance. It fills the gap between spherical and conventional tokamaks, which have high B at high A. The main research topics foreseen are the features of the confinement at high B and low A, Advanced Tokamak regimes, steady-state operation, turbulence and confinement studies with an emphasis on the role of the electric field in confinement, studies of the effects of Zonal flows on transport and confinement (including plasma self-organization, plasma profile resiliency, influence of the q-profile etc.), investigations of MHD effects and disruptions, and Alfven Eigenmode and fast particle studies.

Id 621

Abstract Final Nr. P2.013

Disruption studies and simulations for the development of the DEMO Physics Basis

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In the development of the DEMO Physics Basis an important role is played by the prediction of the plasma disruption features and by evaluation of the EM and thermal loads associated with these events. The development of the DEMO operation points and the design of vessel and in-vessel components are indeed driven by the kind and number of foreseen plasma disruptions and the EM loads produced by these events are by far the largest among those in-vessel components must withstand. A strong integration of physics and engineering expertise is then required to effectively accomplish these studies. These studies include the characterization of the foreseen macroscopic plasma dynamics, with the estimation, based on the experimental data from existing machines, of the range of expected variation for the main parameters characterizing the disruptions: thermal and current quench time, evolution of plasma current, β and li, safety factor limits, halo current fraction and width, radiated heat fraction. The MAXFEA axysimmetric 2D MHD code is used in this work to evaluate the effects on the induced currents and EM loads due to the variation of the disruption parameters and to scan the depandance of the loads from the aspect ratio and plasma currents, analyzing various design options obtained by the PROCESS code. Further, the detailed evolution of the plasma is simulated using the CarMa0NL code, able to couple self-consistently a nonlinear plasma axisymmetric evolution with volumetric 3D conductors, allowing then the evaluation of the effects due to the non-axisymmetric components of the machine as the large vessel ports. The results of these analyses, performed for the worst expected plasma Major Disruption and Vertical Displacement Event, will be used as input for the system level analysis and design of the vessel and relevant in-vessel components.

Id 980

Abstract Final Nr. P2.014

Experiments with heated elongated plasmas with the actively water cooled liquid lithium limiter in view of the development of X-point magnetic configurations on FTU

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Elongated plasmas heated by ECRH up to 650 kW were obtained with a new actively water cooled liquid lithium limiter. These experiments showed the capabilities of liquid lithium capillary-pore systems to withstand continuous large heat loads as required in steady state reactors. The 200 °C water cooling of this limiter allows an efficient control of the lithium melting and the active removal of the heat up to 10 MW/m2. These elongated plasmas have X-point just outside the vessel and flux surfaces opened on the lithium limiter while the LCMS touches the Mo poloidal limiter: anyway, even with the liquid lithium limiter very close to the plasma neither limiter damage nor plasma pollution were observed. A possible access to H-mode, not observed with 200 kA plasma at 5.5 T toroidal field, could be obtained at 2.7 T, where the L-H power threshold is lower than available ECRH power, allowing then the study of the impact of Edge Localized Modes on the limiter. An alternative connection scheme for the poloidal field coils in FTU has been preliminarily analyzed, with the aim of achieving a true X-point configuration with a magnetic single null well inside the plasma chamber and strike points on the lithium limiter. The hardware upgrade required to make this configuration possible includes new busbars through the cryostat to independently supply some of the transformer and feedback coils, a new booster power supply, the reconnection of the passive stabilizing coils in a saddle configuration to increase the vertical instability growth-time. X-point plasma scenarios with current up to 300 kA and duration up to 2.5 s were designed and a first engineering analysis was carried out, showing the structural, thermal and electro-magnetic compatibility of this alternative connection scheme with the load assembly structure and with the existing power supply and cooling plants.

Id 992

Abstract Final Nr. P2.015

Mirror station for studies of the protection of diagnostic mirrors from impurity contamination in ITER: design and first results

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Optical and laser-based diagnostics in ITER will use mirrors to transmit the plasma radiation to the detectors and cameras. The mirror surfaces will be sputtered by the fast plasma particles and contaminated by impurities leading to the degradation of the reflectivity and hampering the performance of the respective diagnostic. Dedicated measures were developed in order to minimize the impurity deposition. These measures envisage the use of shutters and the shaping of diagnostic ducts by using fins to trap impurities on their way towards the mirror located at the end of these ducts. The effect of shaping of diagnostic duct on impurity deposition was studied experimentally in the LHD stellarator. Meanwhile, an assessment of erosion and deposition processes in the ducts with different geometry, conicity and length was performed with a modified B2-EIRENE code. The modeling results predict at least 7-fold suppression of the deposition for the duct having four fins located at the distance of a half of a diameter from each other. The mirror station (MS) was designed to validate the modeling predictions and to study the suppression of deposition inside of diagnostic ducts. The MS contained cylindrical and cone-shaped tubes of different lengths with smooth and shaped geometry of ducts. The ducts of the same length were placed side by side to ensure the same exposure conditions and to allow for the direct comparison of both geometries. The single-crystal molybdenum mirrors were installed in the end of each tube to study the effect of deposition on their reflectivity. The MS was exposed in the midplane port of TEXTOR for about 140 plasma hours. After exposure, no drastic suppression of deposition was observed in the shaped ducts of cylindrical tubes. In the conical tubes no deposition was detected using color fringe analyses outlining the advantages of a cone form. The details of the mirror station along with analyses of the impurity transport in the diagnostic ducts will be presented and discussed.

Id 788

Abstract Final Nr. P2.016

Computational Fluid Dynamics Analysis of the Gaseous Helium Discharge into the Storage Vessel following JT-60SA superconductive Coil Fast Discharge

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Following a fast discharge of the superconducting toroidal field coils of JT-60SA, the coil casing and the winding packs temperature will increase by eddy current heating and the supercritical helium will be expelled from the cooling circuits. The helium pressure in the coils will increase until the safety relief valves will open at 1.6 MPa and the cold helium will be discharged through a 100 m long quench pipe into a 250 m3 storage vessel placed outside the Torus building. Previous analyses had estimated the amount of heat released during the fast discharge and the cold helium mass flow at the inlet of the quench line. The discharge event will discharge a total of about 250 kg of helium at 15 K. The peak mass flow of cold helium at the inlet of the quench tube is estimated to be about 1.2 kg/s and will be over 1 kg/s for the first 90 s before decreasing exponentially to negligible values at 1000 s. Computational fluid dynamic analyses have been performed to assess the impact of the cold gas injection on the quench line and the GHe storage vessel. The analyses have been performed using ANSYS CFX code. For the quench line a simplified model has been used considering only the transient turbulent flow along the pipe. For the storage vessels buoyancy effects have been taken into account in order to estimate local effects in the temperature evolution of the vessel walls. The paper presents the results of the analyses and their impact on the quench line and storage vessel design.

Id 863

Abstract Final Nr. P2.017

The Engineering Design Evolution of IFMIF: from CDR to EDA Phase

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The International Fusion Materials Irradiation Facility (IFMIF), presently in its Engineering Design and Engineering Validation Activities (EVEDA) phase, started in 2007 under the framework of the Broader Approach (BA) Agreement between Japanese Government and EURATOM. The mandate assigned was to develop an integrated engineering design of IFMIF together with accompanying sub-projects to validate the major technological challenges that included the construction of either full scale prototypes or cleverly devised scaled down facilities, which are essential to reliably face the construction of IFMIF on schedule and cost [1]. The engineering design report has released on schedule in 2013 with the Intermediate IFMIF Engineering Design Report (IIEDR) [2] compliant with our mandate. This paper highlights the design improvements implemented from the previous Conceptual Design Phase [3]. [1] J. Knaster et al., IFMIF: overview of the validation activities, Nucl. Fusion 53 (2013) 116001 [2] J. Knaster et al., The accomplishment of the engineering design activities of IFMIF/EVEDA the European-Japanese project towards a Li(d,xn) fusion relevant neutron source, IAEA 2014 [3] IFMIF International Team, “IFMIF Comprehensive Design Report”, International Energy Agency (Jan. 2004)

Id 478

Abstract Final Nr. P2.018

A high frequency, high power CARM proposal for the DEMO ECRH system

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ECRH&CD systems are extensively used on tokamak plasmas due to their capability of highly tailored power deposition, allowing very localised heating and non-inductive current drive, useful for MHD and profiles control. Because of the high electron temperatures expected in DEMO, ECRH systems with operating frequency in the 200 – 300 GHz range, equipped with a reasonable number of high power (Pout ≥ 1MW) CW RF sources, are required for allowing central RF power deposition. Promising alternatives to gyrotrons, having combined limitations in frequency and power, are the Cyclotron Auto-Resonance Masers (CARMs), sources of coherent electromagnetic radiation based on the cyclotron resonance interaction between relativistic electron beams and high frequency TEmn fields in a Bragg resonator with Q ≥ 8000. The high efficiency energy transfer between electron beams and electromagnetic fields is assured by beam velocity spreads less than 0.5%. Operational frequencies γ^2 times higher than the relativistic cyclotron frequency ΩC (γ = relativistic factor) are generated by Doppler shift in the resonator region. ENEA is proposing, on the long term, the design, realisation and test of a high power (P ≥ 500kW CW), high frequency (250 – 300 GHz) CARM able to match the DEMO foreseen plasma characteristics. Two main steps are planned on the short-medium term (=5 years). The first step covers the development and test on dummy load of a CARM with output power up to 100kW and pulse length up to 100µs aimed at demonstrating the feasibility of the source. In the second step a pre-prototype with output power P ≥ 100kW and pulse length t ≥ 100ms will be developed and possibly tested on FTU, which magnetic field (8 T) and plasma characteristics should allow effective test at first and second harmonic. The paper summarizes the results of the preliminary analyses concerning both the CARM and its ancillary systems.

Id 445

Abstract Final Nr. P2.019

Magnetic and thermo-structural design optimization of the Plasma Grid for the MITICA Neutral Beam Injector

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MITICA is a prototype of the Heating Neutral Beam (HNB) Injectors for ITER, with the purpose of validating the injector design and optimizing its operation. Its goal is to produce a focused beam of neutral particles (H- or D-) with energy up to 1 MeV and power of 16 MW for 1 hour. MITICA includes an RF Plasma Source for the production of Negative Ions, a multi-stage Electrostatic Accelerator (up to 1 MV and 40 A), a Neutralizer and a Residual Ion Dump. The magnetic field configuration in the injector is crucial for obtaining the required efficiency and beam optics. A suitable magnetic filter field (FF) is produced by the current flowing through the plasma grid (PG) and related bus-bars. In fact, due to the wide extraction area, it's not possible to obtain a sufficiently uniform FF by means of permanent magnets located on the source edges. The magnetic field distribution in the ion source and accelerator, and in particular the relative strength in the two regions, have been optimized on the basis of the ITER HNB operational requirements and of the experimental results obtained in RF-driven negative ion sources. An optimal configuration was defined by the use of an automated design procedure for inverse electromagnetic problems, in combination with a simplified model of the PG, which assumed a uniform current distribution in the PG. The PG current distribution and the uniformity of the resulting magnetic field have been studied by detailed finite element (FEM) models. The introduction of hollow volumes in the thick copper part of the PG among beamlet groups allows a more uniform PG current distribution and thus a more uniform magnetic field in front of the grid. The paper describes in detail the PG geometry optimization procedure and in particular the related magnetic and thermo-structural FEM analyses.

Id 350

Abstract Final Nr. P2.020

Electrical and structural R&D activities on high voltage DC solid insulators in vacuum

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This paper describes the R&D work performed in support of the design of the ceramic (alumina) insulators for the MITICA Neutral Beam Injector. The ceramic insulators are a key part of the 1 MV electrostatic accelerator of the MITICA injector, as they are required to withstand both the mechanical loads due to the weight of the ion source and the high electric field between the accelerator grids. For this reason, the insulators are a critical element both from the structural and electrical point of view. Numerical Finite Element models have been adopted to study both the electrostatic and the structural problems respectively to calculate the electrostatic field and the distribution of the stress tensor. The numerical models have been benchmarked by experimental tests on samples and insulator prototypes, the experimental campaigns have concerned withstanding voltage tests and destructive structural tests on ceramic samples. The final insulator design has been optimized considering the different requirements coming from structural aspects and voltage withstand issues.

Id 389

Abstract Final Nr. P2.021

The 100kV Faraday cage (High Voltage Deck) for the SPIDER experiment

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In order to optimize the design and operation of the Ion Source for the ITER Neutral Beam Injector (NBI), a dedicated 100 keV Ion Source Test facility, identified as Source for the Production of Ions of Deuterium Extracted from RF plasma (SPIDER), is under construction in the Neutral Beam Test Facility, at the Consorzio RFX premises, in Padua, Italy. The Ion Source, polarized at 100 kVdc Power Supply (100KV PS) negative voltage, is meant to produce negative ions (Deuterium D- or Hydrogen H-) which, after being extracted by the extraction grid, are accelerated up to the ground potential. The required Ion Source and the Extraction Grid Power Supplies (ISEPS) system and the associated diagnostics need to be hosted inside a High Voltage Deck (HVD), a -100 kVdc air-insulated Faraday cage, while a High Voltage Transmission Line (TL) transmits the power and signal conductors from the ISEPS to the Ion Source. The HVD (its procurement started mid 2013 and foresees its delivery on site in the second half of 2014) will consist of a wide mechanical structure (13m (L) x 11m (W) x 5m (H)), designed to support the weight of the ISEPS components, mounted on supporting insulators and clad with a conductive metal sheet in order to reduce the electromagnetic interference (EMI). The paper will report on the design solutions of the HVD and the related progress in the execution of the procurement activities, focusing on insulation, mechanical and thermal issues. The details of the HVD interfaces with the TL, with the SPIDER Cooling Systems and with the 100kV PS are also described. Finite Element (FE) analyses have been performed, on the one hand, to verify the configuration from the electrostatic point of view and, on the other, to evaluate electrical screen effectiveness from the EMI point of view.

Id 404

Abstract Final Nr. P2.022

Control, Protection and Breakdown Management in the Acceleration Grid Power Supply of the ITER Heating Neutral Beam Injector

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The ITER Heating Neutral Beam Injector (HNB) is designed to deliver 16.5MW of additional power to the plasma with pulse duration of one hour. The beam acceleration is obtained by a multistage accelerator composed of five acceleration grids that are powered by the so-called Acceleration Grid Power Supply (AGPS). The AGPS feeds around 56MW at 1MV dc to the acceleration grids in quasi-steady state and is able to interrupt the power delivery in some tens of microseconds in case of grid breakdown. The operation of the AGPS will be controlled by a dedicated control system, the AGPS Local Control (AGPS-LC). The AGPS-LC will be in charge of coordinating the operation and interlocks of the AGPS, putting in place all necessary actions to protect the AGPS from internal and external faults and limit the amount of energy flowing from the power supply to the grids in case of breakdowns. The AGPS-LC will be interfaced to a central control devoted to the operation of the Neutral Beam Injector and to a central interlock system in charge of coordinating interlocks and protection functions of the overall plant. The coordination of actions and their time synchronization is of paramount importance to guarantee a reliable operation. The complexity of the system and the different operational scenarios require an in-depth analysis to identify the most suitable control and protection strategy during faults and breakdowns. Such analysis is based on the existing experience of the JT-60U negative ion HNB system and on dedicated simulations performed on a detailed model of the power supplies. This paper will present a strategy for the coordination of the AGPS integrated protection during faults and breakdowns. The operational scenario has been simulated by using software tools for circuit analysis and a proposal for the design of the AGPS-LC has been derived.

Id 589

Abstract Final Nr. P2.023

Design, manufacture and factory testing of the Ion Source and Extraction Power Supplies for the SPIDER experiment

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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. 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 detailed design of SPIDER ISEPS was finalised in 2011. Manufacture of most components was completed by end 2013 and the Factory Acceptance Tests (FAT) took place early 2014. 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 1 MHz radiofrequency generators of 200 kW output power. Both high voltage, down to -12 kV and high current, up to 5kA, power supplies are present. SPIDER ISEPS will be installed in the NBTF SPIDER High Voltage (HV) Hall, on an air-insulated platform (“HV Deck”), at a nominal voltage to ground of -100kVdc. The limited space available within the HV Deck makes optimisation of the layout a critical aspect. The paper presents the main features of the detailed design developed by OCEM, focussing in particular on the high output voltage pulse step modulators, the high output current resonant converters, the radiofrequency generators by HIMMELWERK GmbH and the architecture and implementation of the complex control system coordinating the individual power supplies. Development and testing of prototypes for power conversion modules and control system customised electronics are also covered. Details are given on non-standard FAT verifying the insulation requirements specific to this application. Performance of ISEPS during the FAT is described and commented against the specifications, with emphasis on demonstration of the load protection requirements, a crucial point for all NB power supplies. Finally, key dates of SPIDER ISEPS installation and site testing schedule are provided.

Id 593

Abstract Final Nr. P2.024

Design and R&D for manufacturing the Beamline Components for MITICA and ITER HNB

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The design of the beamline components of MITICA, the full prototype of the ITER neutral beam injectors, is almost finalised and technical specifications for the procurement are under preparation. These components are the gas Neutraliser, the electrostatic Residual Ion Dump, and the Calorimeter. Electron Dump panels are foreseen sideways at the beam entrance section of the Neutraliser to protect the cryo-panels from electrons stripped in the 1MeV accelerator. High heat fluxes, up to 15MW/m2, with enhanced heat transfer in subcooled boiling conditions will occur in the actively cooled CuCr1Zr panel elements provided with twisted tapes as turbulence promoters. These severe heating conditions will be applied steadily, as the maximum pulse duration is one hour, and cyclically so requiring to fulfil fatigue, creep, and ageing verifications. The design of the beamline components implements and improves technical solutions already tested: the neutralisation gas will expand through injection nozzles located at three alternative positions along the gas cell to look for an optimised neutralisation efficiency, the electric field deflecting the residual ions will be realised by applying the biasing voltage of -20kV added to a trapezoidal waveform of +/-5kV at the frequency of 50Hz in order to spread the power and to increase the fatigue life of the panels, the ex-vessel pneumatic actuator with the double barrier vacuum feedthrough will close the Calorimeter panels for conditioning and commissioning in ITER and for testing of the actuation mechanism in MITICA. Special R&D activities have been undertaken to support the design: manufacturing of thick twisted tapes leading to an increased cooling performance while maintaining flow rate requirements, bending of CuCr1Zr swirl tubes, verification for permanent deformations due to stress relaxation after heating of CuCr1Zr swirl tubes, double side deep drilling of 2m long CuCr1Zr plates. Alignment requirements and remote handling compatibility have been verified.

Id 724

Abstract Final Nr. P2.026

Control and Data Acquisition of the ITER Full-scale Ion Source for the Neutral Beam Test Facility

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The Neutral Beam Test Facility, which is under construction in Padova, Italy, is developing the ITER full-scale ion source for the ITER heating neutral beam injectors (HNB), referred to as the SPIDER experiment, and the full-size prototype injector, referred to as MITICA. The SPIDER control and data acquisition system (CODAS) has been developed and its construction will start in 2014. Slow control and data acquisition will be based on the ITER CODAC Core System software suite that has been designed to facilitate the integration of ITER plant systems with CODAC. Fast control and data acquisition will use solutions specific to the test facility, as the corresponding concepts are not fully mature in the ITER design. The ITER hardware catalogue for fast control has been taken into consideration. The software development will be based on the integration of MDSplus and MARTe, two framework software packages that are well known in the fusion community, targeting data organization and fast realtime control, respectively. The paper will revise the system requirements and the system design, and will show the results already achieved in terms of system integration. In addition, the paper will report the experience in the usage of different cooperating software frameworks and in the integration of industrial procured plant systems.

Id 1027

Abstract Final Nr. P2.027

Upgrade of the TCV tokamak, first phase: neutral beam heating system

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The TCV tokamak [1] contributes to physics understanding of fusion plasmas based on an extensive use of existing main experimental tools: flexible shaping and high power real time-controllable electron cyclotron heating (ECH) system. Improving understanding and control capabilities of burning plasmas is a major scientific challenge, requiring access to plasma regimes and configurations with high normalized plasma pressure, a wide range of temperature ratios, including Te/Ti~1, significant populations of fast ions and relatively low collisionality. These conditions will be reached by an upgraded TCV heating system [2]: installing a neutral beam for direct ion heating and increasing the ECH power injected in X-mode at the third harmonic. The design choices for the NBI installation on TCV were based on consideration of beam access (for which significant modifications of the vacuum vessel are needed), shine through and orbit losses [3]. Detailed studies confirmed the feasibility tangentially injecting a 1MW beam through a port with an aperture of 170×220 mm2. The design of the neutral beam with an energy range of 20-35 keV, tunable power up to 1 MW, and 1-2 s duration is based on a development of the NBI for plasma heating at Budker INP [4]. In order to focus the beam inside the TCV port, the geometry of the grid elementary cell was optimized to minimize the beam's angular divergence. Both beam energy and current are varied during a pulse in order to keep the beam perveance optimal for minimal angular divergence. The installation of the first NB on the TCV is scheduled for 2015. The upgrade of the ECH system foresees (1) the installation of two dual-frequency gyrotrons emitting at 126 GHz (X3, 1 MW/2s/each) or 84 GHz (X2, 0.75 MW/2s/each ECH/ECCD) and (2) replacing two legacy X2/82.7 GHz/450 kW gyrotrons with a units of 750kW capability. This work was supported in part by the Swiss National Science Foundation. [1] A. Fasoli, Nucl. Fusion, 49, 104005 (2009) [2] A. Fasoli et al., Proc. 40th EPS Conf. on Plasma Phys. ECA Vol.37D, P.2.104 (2013) [3] A. N. Karpushov et al., 36th EPS Conf. on Plasma Phys. Sofia, ECA Vol.33E, P-2.140 (2009). [4] A. Sorokin et al., Rev. Sci. Instrum. 81, 02B108 (2010)

Id 53

Abstract Final Nr. P2.028

Upgrade of a 30 kV/10 mA anode power supply for triode type gyrotron

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The RF power of a gyrotron with a triode type magnetron-injection-gun (MIG) can be directly controlled via the voltage applied between its anode and its cathode. Hence, the performance of this type of gyrotron relies directly on the possibilities offered by the power supply controlling the anode to cathode voltage. For a system of gyrotrons connected to the same main high-voltage power supply, with a triode MIG one has the additional advantage of independently controlling each individual gyrotron. This paper presents the modifications brought to the three existing 30 kV/10 mA anode power supplies connected to the 500kW/118GHz/2s X3 gyrotrons operated on the TCV Tokamak. The new working principle is described in detail. Experimental results obtained first on dummy load and then on the gyrotron are compared to simulations performed during the design phase. With respect to the initial working principle [1], the modulation frequency capability has been increased by a factor 10 reaching more than 5 kHz, whereas the output voltage ripple as well as the overshoot/undershoot have been significantly reduced. These improvements of the anode-voltage quality lead to a higher reliability of the gyrotron operation, also allowing us to work closer to the conditions for reaching the maximum RF output power deliverable by the gyrotron. Furthermore, thanks to the increased bandwidth of the modulator, it will be possible to tune, in real time, the gyrotron RF power by varying synchronously the anode and cathode voltages. This power supply upgrade will thereby extend the operation range of the TCV's X3 gyrotrons in view of the forthcoming neutral beam heating installation. [1] D. Fasel et al., Fusion Engineering and Design vol. 56–57, October 2001, pp 633–637

Id 501

Abstract Final Nr. P2.029

Supply equipment to the new NBH system for the TCV tokamak

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The TCV tokamak [1] contributes to the physics understanding of fusion plasmas by investigations based on an extensive use of the following experimental tools: flexible plasma shaping based on 16 shaping coils, and high power real time-controllable electron cyclotron heating (ECH) system, made of 9 gyrotrons capable of 500kW RF power each (6 at the 2nd harmonic, 81GHz; 3 at the third harmonic, 118GHz) [2][3]. One of the major scientific challenge, the TCV upgrades aimed to, will be the access to a wider range of temperature ratios up to Te/Ti~1. For this purpose, the TCV heating system will be improved by the integration of a neutral beam heating (NBH) system impacting directly on the ion temperature (Ti), while gyrotrons are used to vary mainly the electron temperature (Te). This NB injector will provide 1MW of neutral power during 2sec into the TCV plasma, at a nominal energy of 30keV, for an electrical power installed of 2.2MVA. This paper will focus on the requirements imposed for installing and commissioning the NBH system on CRPP site. Looking first at the electrical supply, the operation of the NB implies to connect the HVPS either to the local grid (20kV, 50Hz) for conditioning purpose, or to the MG delivering 10kV at 120Hz during the TCV pulses. The conditions leading to use these two power sources with the NBH system will be described. The criteria imposing the final layout chosen for the power part, which will integrate the safety rules considered for the equipment as much as for the humans, will be developed. An overview on the injector supply system (HV, RF, auxiliaries) will be also provided, pointing out the main choices leading to the final design. Then, the cooling system allowing to operate the NB equipment in conditioning mode (under a calorimeter) or into plasma pulses will be detailed. The design criteria, taking care of the duty cycle for which the injector installation is performed, will be highlighted. The integration of the NBH installation into the TCV operation safety system will conclude this overview on the preparation work to supply optimally the novel NBH injector for TCV. This work was supported in part by the Swiss National Science Foundation. [1] A. Fasoli, Nucl. Fusion, 49, 104005 (2009) [2] D. Fasel et al., Proc. of 19th SOFT Conf. in Lisbon, 16-20.09.1996, Vol.1, p.569-572 [3] T.Goodman and TCV Team, Nucl. Fusion, 48, 054011 (2008)

Id 547

Abstract Final Nr. P2.030

Prototyping and tests of ITER ECH waveguide components for the first confinement system

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The EC H&CD Upper Launcher (UL) is to be used for the control of MHD activity by focusing up to eight mm-wave beams. The four EC Upper Launchers are divided into two european procurement arrangements (PA), with PA1 comprising the first confinement system (FCS) mainly located in the port cell, and PA2 comprising the port plug and the in-vessel mm-wave components. The FCS, a level 1 system according to the ITER Safety Important Classification Criteria (SIC-1), consists of the diamond window, the down-taper, the isolation valve, mitre bend, straight waveguide components, a port plug closure plate and feed through. Linking the waveguides routed in the tokamak building to the quasi-optical launcher antennas, the FCS shall guarantee vacuum integrity under various torus operating conditions prevailing during assembly at cold and/or working temperature, baking, plasma startup and burn phases, off-normal events, such as loss of cooling, hydrogen detonation, or fire in the port cell area. The structural and thermo-mechanical loads on the FCS components, mainly determined by the thermal expansion of the ITER vacuum vessel relative to the building, result in specific design choices, validated by a prototyping and test programme for a subset of waveguide components, such as straight waveguide segments connected by the newly designed waveguide coupling systems including double metallic seal arrangements with differential vacuum testing, and a new type of monolythic mitre bend. Structural and thermal loads are applied while vacuum is tested in the line. The corrugated waveguides have diameters of 50 mm and 63.5 mm compatible with the transmission of 1.5 MW per beam at 170 GHz. This work was supported in parts by the Swiss National Science Foundation and by Fusion for Energy under Grant F4E-GRT-161 and within the Electron Cyclotron Heating Upper Launcher Consortium (ECHUL).

Id 353

Abstract Final Nr. P2.031

In-situ upgrade of the TCV vacuum vessel with tangential ports for the neutral beam plasma heating system

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The Tokamak à Configuration Variable (TCV), built between 1989 and 1992 at EPFL is the object of a major revision for the installation of a 1 MW neutral beam heating (NBH) injector1. For efficient plasma heating the NBH injector requires a specific geometric arrangement, with the beam line at mid plane oriented tangentially relative to the plasma axis and a port with rectangular aperture. The actual TCV vacuum vessel (VV)2, built between 1989 and 1991 at the production site of De Pretto Industrie (DPI), does however not offer adequate ports. Furthermore, the location of numerous components surrounding the TCV VV are preventing the NBH injector integration. The acquisition of a completely new VV would have entailed the destructive removal of many TCV components (mainly the toroidal and poloidal field coils), high costs of the upgrade and delayed the physics campaigns. Therefore, the in-situ modification has been preferred. It consists in adding two new tangential ports in the VV and transforming existing surrounding components, such as the adjacent ports, the coil support structures, the graphite first wall, the external thermal insulation, the heating pipes, the magnetic flux and other diagnostic loops. The extremely confined space surrounding the location of the new tangential ports renders the project challenging. The feasibility study and the in-situ modification of the VV is contracted out to the original VV manufacturer DPI. This paper describes the design and analysis, the modification and the recommissioning of the TCV VV to adapt it for the integration of the NBH injector. This work was supported in part by the Swiss National Science Foundation. [1] A. N. Karpushov et al., Upgrade of the TCV tokamak, first phase: neutral beam heating system. This conference. [2] C. Hollenstein et al., The TCV Vacuum Vessel : Design and Construction. 17th Symposium on Fusion Technology, Rome, Italy, 1:282-286, September 1992.

Id 719

Abstract Final Nr. P2.033

User requirements and conceptual design of the ITER Electron Cyclotron Control System.

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The ITER Electron Cyclotron plant is a complex system, essential for plasma operation. The system is being designed to supply up to 20MW of power at 170 GHz; it consists of 24 RF sources connected by switchable transmission lines to four upper and one equatorial launcher. The complexity of the EC plant requires a Main Controller, which provides the functional and operational interface with CODAC and the Plasma Control System and coordinates the various Subsystem Control Units, i.e. the local controllers of power supplies, gyrotrons, transmission lines and launchers. A conceptual design of the Electron Cyclotron Control System (ECCS) was developed, starting from a collection of the user requirements organised as a set of operational scenarios exploiting the EC system. The design consists in a thorough functional analysis, including also protection functions, and in the development of a conceptual architecture for both software and hardware. The main aim of the work was to identify the physics requirements and to translate them into control system requirements, in order to define the interfaces within the components of the ECCS. The definition of these interfaces is urgent because some of the subsystems are already in an advanced design phase. The present paper describes both the methodology and the resulting design.

Id 597

Abstract Final Nr. P2.034

An adaptive disruption predictor based on FDI approach for next generation Tokamaks

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Disruptions have the potential to create serious damage to large reactor-scale. Thus, disruption detection is essential to allow proper mitigation actions to be triggered when preventing disruptions is not feasible. Several contributions have been proposed using neural network models with good performances in different tokamaks. In particular, Support Vector Machines in JET [1, 2] and Multi-layer Perceptrons both in JET [3] and ASDEX Upgrade [4]. The main drawback of these methods is the need of a set of disruptions to implement the predictive model. As largely known, ITER cannot wait for hundreds of disruptions to develop a successful disruption predictor. Hence, a prediction system starting from only few safe discharges will be required. Thus, the proposed approaches are not directly applicable. To this purpose, the disruption prediction can be formalized as a fault detection and isolation (FDI) problem, where the safe pulses are assumed as the normal operating condition and the disruptions are assumed as status of fault [5]. The main advantage of the proposed FDI methods is that the model can be developed without any information about disruptions. In this work, in view of ITER, an adaptive disruption predictor based on FDI approach is developed. In particular, an autoregressive model is trained to represent the normal operating conditions (NOC) described by few safe shots. Then, the model is progressively updated as soon as a new safe configuration is performed. The dynamic structure of each pulse is estimated through the fitting of the NOC model, the discrepancy between the outputs provided by the NOC model and the actual measurements (residual) is an indication of the plasma disruptivity. The prediction performance is evaluated for JET using a set of safe and disrupted discharges in terms of correct predictions, missed and false alarms. Preliminary results show the suitability of the proposed method. [1] B. Cannas et al. 2007 Fusion Engineering and Design 82, Issues 5 no. 14, 2007, 1124 – 1130. [2] J. Vega et al 2013 Fusion Engineering and Design 88, 1228-1231. [3] B. Cannas et al 2007 Nucl. Fusion 47, 1559–1569. [4] B. Cannas et al. 2010 Nucl. Fusion 50 075004. [5] R. Aledda et al. 2012 IEEE Trans. On Plasma Science 40, no.3, 570 – 576.

Id 545

Abstract Final Nr. P2.035

Three-dimensional electromagnetic analysis of JT-60SA conducting structures in view of RWM control

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The JT-60SA experiment [1]-[2] is a joint international project between Japan and Europe, aiming at contributing to early realization of fusion energy by supporting the exploitation of ITER and by complementing ITER in resolving key physics and engineering issues for DEMO reactors. One of the design requirements [1] is to sustain a high plasma beta exceeding the no-wall ideal stability limits, which will give rise to MHD instabilities - in particular, Resistive Wall Modes. In order to feedback control these modes, 18 internal stabilizing coils are foreseen, together with a double-wall stabilizing conducting shell. For the dimensioning of the power supplies, a simple 2D analysis of the system may be sufficient [3]. Conversely, a quantification of the effectiveness of the feedback coils and of the actual stabilization properties of the structures requires necessarily more detailes studies. This work presents the three-dimensional electromagnetic numerical analysis of active and passive conducting structures of JT-60SA which are relevant for RWM control. This is carried out with two different codes (CAFE [4] and CARIDDI [5]), in order to gain confidence in the results. The final aim is to get to an overall 3D model of the system which includes the plasma response [6]. This work was partially supported by the Italian MIUR under PRIN grant 2010SPS9B3. [1] S. Ishida et al. , Nucl. Fusion, 51 (2011) 094018 [2] Y. Kamada et al, Nucl. Fusion 51 (2011) 073011 [3] A. Ferro et al, Fus. Eng and Des., 88 (2013) 1509 [4] P. Bettini et al., IEEE Trans. on Magnetics, 50 (2014), 7000904 [5] R. Albanese, G. Rubinacci, IEE Proc., 135A, 5 (1988) 457 [6] F. Villone et al, Phys. Rev. Lett. 100 (2008) 255005

Id 609

Abstract Final Nr. P2.036

Improving the performance of the JET Shape Controller

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The JET Shape Controller (SC) is part of the Plasma Position and Current Control system (PPCC). SC uses nine distinct circuits, energizing the JET poloidal field coils, to control in real time the coil currents and the plasma parameters such as plasma shape, current and position. The control scheme presently used [1] is based on a Multiple Input Multiple Output (MIMO) controller, which is designed to decouple the mutual coupling of the different coils. Achieving such a decoupling, the SC allows the user to independently tune the time response of each circuit. As a matter of fact the intended decoupling algorithm has been incorrectly coded in the JET SC system; however the controller parameters have been tweaked during an extensive commissioning phase back in 1997, in order to obtain a good overall control. This paper describes the modelling and experimental activities performed to correct the code error and the related empirically optimised control parameters, and to validate and test an improved set of SC parameters, obtained by using the CREATE model, which has been extensively validated at JET [2-3]. The same reliable model allowed to perform simulations to optimise the SC in specific control modes. Firstly the simulation scheme used to reproduce the plant behaviour is presented. The use of this tool minimized the machine time required for (re-)commissioning. Secondly the design procedure used to correctly decouple the PF currents in closed-loop and to increase the performance is described. Finally the commissioning activities carried at JET in 2013 and 2014 are presented. [1] M. Garribba et al., Proceedings 15th SOFE Conference, Massachusetts, 1993, pp. 33-36. [2] R. Albanese et al., Fusion Engineering and Design, vol. 74, no. 1-4, pp. 627-632, 2005. [3] R. Albanese et al., Fusion Engineering and Design, vol. 86, no. 6-8, pp. 1030-1033, 2011.

Id 560

Abstract Final Nr. P2.037

From Use Cases of the Joint European Torus towards Integrated Commissioning Requirements of the ITER Tokamak

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The Joint European Torus (JET) is the largest tokamak currently in operation in the world. One of the greatest challenges of JET is the integrated commissioning of all its major plants. This is driven, partially, by the size and complexity of its operational infrastructure and also by the fact that, being an international environment, it has to address the issues of integrating, commissioning and maintaining plant systems developed by third parties. The ITER tokamak, now in construction, is a fusion device twice the size of JET and, being a joint effort between the European Union, China, India, Japan, South Korea, the Russian Federation and the USA, it will share on a wider scale all of the JET challenges regarding integration and integrated commissioning of very large and complex plant systems. With the scope of leveraging and contributing with some of the history and experience of JET into the ITER project, Fusion for Energy (F4E) has worked together with the Culham Centre for Fusion Energy (CCFE), the host and operator of JET, for the provision of user experiences related to the integrated commissioning of the tokamak. Since the collection of commissioning experience from a tokamak is a very broad subject, the first stage of the work has focused mostly on the commissioning of the plant systems' hardware and software and its impact on the Control and Data Acquisition System infrastructure, in particular, the real-time framework. In particular, F4E has collected, processed and structured details about design choices that have proven successful, technical mistakes and lessons learned, risk assessment, changes to the design, and conflicts between plant system commissioning requirements and how these were addressed This work presents and discusses in detail the main results and the methods that were used to extract and translate the commissioning experience information into ITER requirements.

Id 418

Abstract Final Nr. P2.038

The ITER plasma control system simulation platform

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The Plasma Control System Simulation Platform (PCSSP) is a highly flexible, modular, time-dependent simulation environment developed primarily to support development of the ITER Plasma Control System (PCS). It has been under development since 2011 and is scheduled for first release to users in the ITER Organization (IO) and at selected additional sites in early 2015. Modules presently implemented in the PCSSP enable exploration of axisymmetric evolution and control, basic kinetic control, and tearing mode suppression. A basic capability for generation of control-relevant events is included, enabling study of exception handling in the PCS, continuous controllers, and PCS architecture. While the control design focus of PCSSP applications tends to require only a moderate level of accuracy and complexity in modules, PCSSP is also capable of embedding or connecting to more complex codes to access higher accuracy if needed. This paper describes the background and motivation for PCSSP, provides an overview of the capabilities, architecture, and features of PCSSP, and discusses details of the PCSSP vision and its intended goals and application. Completed work, including architectural design, prototype implementation, reference documents, and IO demonstration of PCSSP, is summarized and example uses of PCSSP are provided. The high level objectives of ongoing work in 2014 are summarized and include preparation for release of a 'beta' version of PCSSP for IO users and distribution to selected laboratories worldwide and preparation for the next phase of development of PCSSP. High level objectives for PCSSP work beyond 2014 are also discussed. Work was supported by the ITER Organization under ITER/CTS/6000000037.

Id 658

Abstract Final Nr. P2.040

Design and commissioning of controller for power supplies for vertical position control on the COMPASS tokamak

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The vertical plasma position on COMPASS tokamak is normally controlled by a feedback power supply, called fast amplifier for radial magnetic field (FABR), which is based on MOSFET components and capable to modify current in the control coils by approx. 1 kA/ms. Recently, a new power supply for fast vertical movements of the plasma column was put in operation, so-called vertical kicks power supply (VKPS). VKPS was designed for experiments dedicated to ELM triggering by fast change of the plasma position, however, it can be used also for disruption studies: either to slow down a disruption or – vice versa – to induce a disruption. It is based on IGBT modules and can assure a current change up to 8 kA/ms in the control coils. Both the VKPS and FABR are connected serially to the vertical position control coils. To ensure a coordinated operation of the both power supplies, a common controller was needed to control each of the power supplies individually as well as to prevent any undesirable interaction between them. Such a controller was designed and manufactured and it is regularly used on COMPASS at present. The controller assures the regular feedback control as well as fast disturbances - the vertical kicks. A high speed optical link enables efficient and simultaneous configuration of VKPS and FABR from the real-time control system MARTe and decreases transport delay in the control loop. This contribution will describe the control of the two serially connected power supplies. Further, the controller design, software development, simulations of the circuit, and results of test measurements will be presented. Finally, the results of the regular operation will be shown.

Id 993

Abstract Final Nr. P2.041

Plasma density real-time control for the compass tokamak

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The control of plasma density plays an important role in the tokamak operation. In case of the COMPASS tokamak, the density control is performed indirectly by controlling the working gas injection using a piezoelectric valve, which is controlled in a feedback loop. This feedback loop uses measurement of the plasma density obtained by 2 mm microwave interferometer as the input. The interferometer measurement uses two near frequencies 131 GHz and 133 GHz. Newly, an intermediate frequency conversion at 133 MHz is used which allowes us to recover plasma density after the fringe jumps in the real-time control loop.

Real-time control is based on MARTe (Multi-threaded Application Real-Time executer) framework at COMPASS. Two different threads are used to calculate (fast 50 us thread) and to control (slow 500 us thread) the plasma density. The gas puff valve has been calibrated using predefined waveforms and the pressure changes have been measured in the tokamak vessel. Relation between the opening of the valve corresponding to the applied voltage and the pressure inside the tokamak vacuum vessel was implemented inside a dedicated General Application Module (GAM) in the real-time framework MARTe. In this contribution, the HW and SW implementation of the real-time control of plasma density on COMPASS will be presented.

Id 1015

Abstract Final Nr. P2.042

Implementation Strategy for the ITER Plasma Control System

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The ITER Plasma Control System (PCS) is a fundamental component of the ITER Control, Data Access and Communication system (CODAC). It will control the evolution of all machine and plasma parameters that are necessary to operate ITER throughout all phases of the discharge. The ITER CODAC section is responsible for the design of the system architecture of the plasma control system and the implementation of the control strategies and algorithms provided by the ITER Plasma Operation Directorate. This paper will give an overview of the implementation strategy and current state of design of the tools to support the implementation and summarize outcome of design reviews held in late 2012 and early 2014 with focus on the challenges identified and the approach taken by the CODAC section to design a system architecture which is capable of meeting them. The central tool to implement plasma control functionality will be a real-time framework, which is currently under development with strong support of the worldwide fusion community. This framework should not only support the implementation of plasma control strategies with the extensive exception handling and forecasting functionality foreseen for ITER, but also integrated commissioning and sophisticated machine protection functionality (e.g. first wall protection). These are novel requirements not or only partially addressed by existing plasma control systems. The paper will summarize the key challenges and present the implementation strategy until initial ITER operation. A second cornerstone in the implementation strategy is the development of a powerful simulation environment (Plasma Control System Simulation Platform - PCSSP) to design and verify control strategies, event handling and interaction with the ITER Central Interlock System. One objective of this simulation environment will also be to facilitate the efficient implementation of the actual ITER Plasma Control System. The PCSSP environment is currently under contract and this paper will also give an overview of its current state of development.

Id 130

Abstract Final Nr. P2.043

Digital nuclear radiation spectroscopy: Hardware requirements to minimize energy resolution degradation

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Nuclear radiation spectroscopy is now often relying on digital acquisition techniques. The present paper addresses the problem of analyzing systematically the requirements of the digitizer in terms of sampling frequency and number of bits to minimize the energy resolution degradation caused by the digitizing process. The analysis is performed using synthetic pulses (with different amplitude and FWHM) of typical nuclear spectroscopy detectors and the pulse area as energy estimate. Additional relevant issues, such as the hardware architecture and the data throughput speed, are also discussed.

Id 239

Abstract Final Nr. P2.044

Triple modulation radar electronics with improved phase disambiguity

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During the past few years in ENEA Frascati laboratories has been developed a Digital Radar Electronics, efficiently used in systems such as interferometry, reflectometry and optical radars. The system measures the flight time between outgoing and echo signals by using the phase difference measured comparing the TX and RX signals. In general, the systems developed are based on amplitude modulation technique, having single or double-modulation frequency. The double modulation frequency is used effectively to increase the maximum measurable delay time, by solving the ambiguity related to the echo phase wrapping. During 2014, a triple-modulation radar electronics along with an algorithm able to solve the phase disambiguation problem were developed. The aim of the system up-grade was the increase of the robustness in the phase-wrapping disambiguation, in particular in presence of noise. The new triple-modulation radar electronics has been efficiently applied in the IVVS probe prototype, which is a 3D Laser radar developed for hostile environments such as ITER. The triple-modulation radar electronics improves the reconstruction capabilities in the 3D images, especially in the case of low amplitude echo signals, without any degradation in the resolution and range performances for the IVVS probe. In the present document, the updates made on the triple-modulation radar electronics are described, together with the test results obtained by comparing the performance of the ENEA IVVS prototype system that uses double or triple-modulation.

Id 791

Abstract Final Nr. P2.045

Control and data acquisition system with multiple configurations for tests in nuclear facilities

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The In-Vessel Viewing System is a 3D laser scanning system which will be used to inspect the blanket first wall in ITER. To make the IVVS probe design compatible with the harsh environmental conditions present in ITER, a test campaign was carried out in 2012-2013 to verify the adequacy of the main components of the IVVS probe. In particular the IVVS components inspected were two customized ultrasonic piezoceramic motors, one optical encoder and other passive components. Tests were performed in four different ENEA facilities to verify compatibility with: gamma irradiation, neutron irradiation, magnetic field, high vacuum and high temperature. The motors tested were equipped with various sensors (position sensors, switches, torque-meter and temperature sensors); the sensors were selected depending on the specific facility constraints and test requirements. A general architecture of the Data Acquisition and Control System (DACS) was defined and then specialized for each test. To be suitable for this test campaign, the DACS had to host various I/O modules and to properly interface the driver of the customized piezo motors, in order to permit the full control of the test and the acquisition of experimental data, which could be very large in size in long-term tests. The tests were located in different facilities so the DACS system had to be compact and portable; furthermore, since the nuclear facilities had restricted access, the DACS system had to be remotely controllable and maintainable. The Data Acquisition and Control Application was developed in four specialized versions, one for each test, allowing the online control of the test and the presentation of all relevant pre-elaborated data. At the same time, all the full raw data were stored to allow offline analysis. This paper presents in details the DACS developed for the described test campaign, including requirements, architecture design and implementation.

Id 827

Abstract Final Nr. P2.046

First results on runaway electron studies using the FTU neutron camera

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A digital upgrade of the analogue electronics of the Frascati Tokamak Upgrade (FTU) Neutron Camera [1] has been carried out in order to enable studies on runaway electrons (RE) by measuring the hard x-rays (HXR) produced in the bremsstrahlung interactions between RE and plasma ions. The gamma camera system is based on six radial lines of sight equipped with liquid organic scintillators (NE213) capable of n/γ discrimination. The digital acquisition system is composed by three Innovative Integration X65400M digitizers each one with two 14-bit 400MSamples/s ADCs enabling the separation between neutron and hard x-ray events also in conditions of very high count rate (MHz range). First measurement results and correlations with Mirnov coils and FEB data indicate the capability of the diagnostic to provide HXR profiles for energies > 0.1 MeV and information on the runaway population during the Ip ramp-up, flat-top and ramp-down phases with sub-ms time resolution. Disruption events can also be analysed, although in such phases the data are affected (mainly in the inner three detectors) by the presence of a strong HXR background not originating in the plasma. [1] P. Batistoni et al., Rev. Sci. Instrum. 66, 4949 (1995)

Id 988

Abstract Final Nr. P2.047

Double pulse Laser Induced Breakdown Spectroscopy measurements on ITER-like samples

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Laser Induced Breakdown Spectroscopy (LIBS) is an attractive analytical technique for the in situ analysis of the surface layer composition of the components inside the vacuum vessel of ITER and for estimating the tritium content in the machine. On the other hand single pulse (SP) LIBS is less sensitive as compared to other spectrometric methods: to improve its sensitivity a dual pulse (DP) configuration, with two laser pulses with inter-pulse delay in the range of nanosecond to several microseconds can be used. In this work DP measurements in vacuum on samples resembling ITER divertor situation after material re-deposition were carried out jointly by ENEA and IPPLM associations in the frame of EFDA task WP13-IPH-A01-P3-01. Measurements used the 1064 nm laser line of a LOTIS LS-2131 Double Pulse Nd:YAG laser with interpulse delay ranging between ~20 ns – 80 ms and pulse duration of 8-20 ns. The collinear configuration was applied, with the two laser pulses propagating to the sample along the same axis. Samples consisted of tungsten substrate coated with 3μm thick mixed layer of C/Al/W (Al as a proxy for Be). With DP technique an enhancement, with respect to SP LIBS, of the emission lines throughout all the wavelength range detected was found, critically depending on inter-pulse separation and gate delay of the light collection. Calibration Free (CF) method was applied to the spectroscopic data to get the elemental concentration of the sample. Plasma electron temperature and density, needed to be known for applying the CF method, were inferred by using the Boltzmann Plot method and the line Stark broadening, respectively. Despite the not optimal geometry of emitted light collection, the pattern of concentrations reproduces in a satisfactory way the values found by EDX analysis carried out in zones of the samples far from the laser spots.

Id 998

Abstract Final Nr. P2.048

Diagnostic Setup for Investigation of Plasma Wall Interactions at Wendelstein 7-X

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Wendelstein 7-X being the most advanced stellarator is currently commissioned at Greifswald. Forschungszentrum Jülich is preparing a research program in its particular field of expertise – plasma wall interactions (PWI) – by developing a dedicated set of diagnostic systems. While PWI are studied at many fusion devices as well as plasma generators, the specific interest of this research field at Wendelstein 7-X is to understand the processes in presence of a 3D plasma boundary of an island divertor. Furthermore, for the first time steady state plasma at high density and low temperature in the divertor region will be available. Since PWI only could be understood in conjunction with the edge plasma properties the aim of the setup is to observe both the edge plasma as well as the surface processes. For optimum combination of different diagnostic methods the edge diagnostic systems are aligned along one out of the five island chains in the standard magnetic configuration. Main elements are as follows. A multipurpose fast manipulator is to expose Langmuir, magnetic and Mach probes for determination of plasma edge parameters and gas feed, material samples and other objects for specific PWI processes. Two gas boxes in opposite divertor modules allow helium puffing for profile measurements, impurity seeding for edge cooling and transport studies as well as edge fuelling. Two endoscopes each observing the divertor region at the gas inlets are foreseen to detect helium radiation for profile measurements, tomographic reconstruction of impurity radiation, and transient foot print on the divertor via infrared light detection. A poloidal correlation reflectometer will allow studying local pitch angles of the magnetic field and turbulences. VUV and X-ray spectroscopy in the plasma core are to deliver information on impurity accumulation. The concept of the diagnostic setup is presented in this contribution.

Id 413

Abstract Final Nr. P2.049

Design overview of the ITER core CXRS fast shutter and manufacturing implications during the detailed design work

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A fast shutter is one of the subsystems considered in the ITER core charge exchange recombination spectroscopy (CXRS) diagnostic. Its main goals are to protect critical optical components against degradation, mainly the first mirror (M1), and provide means of calibration for the diagnostic optical system. Several ideas have been studied and subjected to changes over the time, according to different core CXRS conceptual baselines. As a continuing task of these previous concepts and studies, a detailed solution was developed and is presented for the current design stage. The shutter's solution is driven by a bidirectional (double action) frictionless helium actuator. The shutter structure consists of two protecting blades with calibration on the top surfaces and integrated cooling channels, two arms interconnected to form one cooling circuit including the blades, a bumper system to limit the arms movement, and a support to hold all other components. Several issues encountered during the detailed design work such as the joining processes, machining limitations, bending limitations, among others, are mentioned in this paper. The respective solutions and parallel ongoing investigations are explained in consideration with system requirements, technological capabilities and know-how of our in-house and industry workshops. Detailed manufacturing drawings have been developed for all shutter components as the deliverable final product of the current design stage. These drawings are being used for prototyping several shutter structures and actuators planned for the next design step which includes testing, numerical benchmarking, and validation of the ITER core CXRS fast shutter concept.

Id 493

Abstract Final Nr. P2.050

Dynamic performance of frictionless fast shutters for ITER: numerical and analytical sensitivity study for the development of a test program

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Fast shutters are designed to prolong the lifetime of ITER diagnostic first mirrors. The concept of an elastic fast shutter that operates frictionless in vacuum has been extensively studied at FZ-Juelich, Germany. The developed shutter represents a relatively long mechanical structure (~2 m) that operates in fractions of a second. Under actuation, two fork-like shutter arms bend between two pairs of limiting bumpers thus blocking the path to the mirror. Due to the fast actuation the arms rebound and oscillate on the bumpers and thus deteriorate mirror protection and measurements. The amplitudes of these movements are mainly determined by the shutter stiffness, its natural frequencies and the time profile of an actuation force. The paper presents parametric numerical studies of the shutter dynamic behavior. A semi-analytical model was developed to predict the shutter impact kinetic energy that mostly determines its further dynamic response. The structure sensitivity to different parameters was studied and ways for shutter optimization were laid down. The first prototype shutter is made in a full-scale complexity (designed for severe ITER environment) to check its manufacturability and achievable tolerances. Another manufactured shutter is a parametric simplified mockup with easily changeable mechanical characteristics like its stiffness and natural frequencies. Basing on the analysis performed a test program aiming at the shutter further optimization has been developed. Both mockups will be tested in the air and vacuum under different actuation methods like a simple mechanical drive or a complex gas actuator. Tests in the vacuum let rule out damping by the air. The powerful capabilities of the parametric mockup will be engaged to optimize the shutter dynamic performance. The numerical models will be validated/corrected by the test results and, in their turn, will help to explain the test results and probably correct the test program. The lifetime of the shutter critical components can be tested in the air for the required number of loading cycles.

Id 614

Abstract Final Nr. P2.051

Major aspects of the design of a first mirror for the ITER core CXRS diagnostics

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The ITER core charge exchange recombination spectroscopy diagnostic (cCXRS) occupies the vacuum vessel upper port #3 and includes, in its generic version, the following in-vessel components: an optical mirror system, a shutter, the diagnostic first wall (DFW) and the neutron shielding block (DSM). The mirror system transmits the visible light produced by the interaction between the plasma and a diagnostic neutral beam (DNB) to the dedicated spectrometers. The shutter blocks\opens the entrance aperture and limits particle fluxes reaching the mirrors. The DSM performs a standard protective function against neutrons and gamma radiation, and carries the main diagnostic components. The most vulnerable diagnostic mirror is obviously the first mirror (M1). The anticipated optical lifetime of the mirror (which, according to different sources, can be limited to several months) is the subject of an intensive R&D programme. The main aim of the proposed design solutions is to identify the most reasonable and reliable M1 structure at present. It is shown that the mirror interface with the DSM and DFW units has a major impact on the M1 design. The applicability of each option is determined by many reasons, and especially, by the ITER generic UPP layout and its customization flexibility.The examined mirror options have some common parts like a single-molybdenum-crystal-mirror (ScMo) and some variable parts like supporting and cooling systems. The largest dimension of the mirror optical face is ~ 300 mm. Such large ScMo workpieces are currently not available on the market. It forces to design the mirror as an assembly of several ScMo pieces joined together. The M1 design is supported by multifield thermal, electromagnetic and structural analyses. The study conducted confirms the feasibility of the proposed solutions. At the same time the paper indicates technological issues of the M1 unit to be solved in future.

Id 613

Abstract Final Nr. P2.052

Aim and features of the simplified parametric mock-up of a fast shutter developed for ITER optical diagnostics

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To optimise the mechanical behaviour of the Fast Shutter intended for ITER Diagnostics, a simplified mock-up was developed and manufactured in addition to the prototype that was designed to conform to the harsh ITER environment. This parametric shutter is to be used for mechanical tests in the air and vacuum conditions at FZJ Juelich, Germany. The main idea of the simplified shutter mock-up was to reduce the structure complexity but to retain the mechanical properties like characteristic dimensions as well as the distribution of mass and stiffness. For this reason, a basic mass-stiffness-equivalent geometry was abstracted into a simple beam structure. The mechanical features that are important for the dynamic performance can vary. The initial "mathematical" model has passed through marginal but influencing design modifications to provide mounting on a support structure, enhancement of positioning flexibility and also to provide manufacturability. The main feature of the simplified shutter is its easily changeable geometry which in turn will change the shutter mechanical (especially dynamic) behaviour. Note that the mock-up simplicity has some drawbacks: for instance, it operates correctly only in the horizontal position, not inclined, a feature to be duly considered during the tests. The paper mainly focuses on numerical studies of possible ways to change by simple means the shutter mass-stiffness distribution thus influencing its mechanical performance according to the developed test programme. Results show that attachment of simple parts on the mock-up is sufficient to change relevant natural frequencies and system stiffness, thus the shutter dynamic can be adjusted as required within the considered range. Furthermore, numerical simulations have been conducted to estimate the impact of manufacturing/assembly tolerances on the mechanical performance of the shutter and to derive requirements on tolerances.

Id 585

Abstract Final Nr. P2.053

Testing of SiO2 / TiO2 coating on a stainless steel substrate under ITER in-port conditions

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A broadband multilayer dielectric coating of TiO2 / SiO2 was manufactured on a stainless steel substrate (1.4429) with a mean specular reflectivity approaching 98% over the wavelength range of 450-670 nm [1]. The coating was tested under thermal conditions as expected for in-port mirrors in ITER up to 61 days at $T_{mirror} \geq 200°C$ and exposed to water vapour atmosphere at 120°C. Reflectivity measurements were conducted at elevated temperature and room temperature in between long-term tests. The mirror coating exhibits no permanent change in specular reflectivity after heating to 350°C, adding to the varied behaviours as found in previous tests [2]. For thermal loads up to 17 days at $T_{mirror} \geq 150°C$ no visible defects were noticed but they cannot be completely ruled out. Local defects developed during further testing in the form of blisters and flakes with up to 200 µm diameter. The blisters grew and flaked off over several days at elevated temperature. Exposure to water vapour and thermal gradients of +100 K/h and -200 K/h did not accelerate subsequent generation or growth of defects. Based on the results, partial coatings with different interlayers were exposed to elevated temperature to study the development of defects. [1] A. Krimmer, G. Kassek, H.J. Allelein, Yu. Krasikov, O. Neubauer, Design and testing of secondary mirrors for the core CXRS diagnostic system in ITER, Fusion Engineering and Design, Volume 88, Issues 9–10, October 2013, Pages 2021-2024 [2] I. Orlovskiy, E. Andreenko, K. Vukolov, T. Mukhammedzyanov, A. Tobengauz, Broadband dielectric mirrors for optical diagnostics in ITER, Fusion Engineering and Design, Volume 88, Issues 6–8, October 2013, Pages 1284-1287

Id 641

Abstract Final Nr. P2.054

Development of radiation hard quartz microbalance for fusion devices

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Control of the tritium inventory in fusion devices is one of most important issues for development of fusion power with acceptable level of environmental hazards. According to the present knowledge, a major part of the inventory is related to the co-deposit layers formed in the remote areas. The monitoring of the co-deposits layers in these areas can provide in-situ measurement of the evolution of this fuel inventory. The quartz crystal microbalance (QMB) method is often successfully used in fusion experiment for investigation of the first wall erosion and deposition processes. Despite of QMB is well-established technique for the measurement of film deposition grow rate in both laboratory and industry, the use of standard equipment based on semiconductor electronics in fusion experiment has several drawbacks. The major drawbacks are the low radiation hardness and high risk of electrostatic damage. These make semiconductor based QMB detector practically inapplicable in large scale fusion experiment dealing with D-T fuel mixture, where quartz crystal sensor electronics should be in the vacuum, close to the plasma edge. Contrary to the semiconductors, electronics based on vacuum tubes are practically insensitive to both neutron and gamma radiation damages. In addition, it can withstand much higher temperatures and overvoltage. Miniature vacuum tube - nuvistor 8085 was chosen for quartz oscillator circuit. It has been proved that the chosen nuvistor was able to sustain high neutron fluxes. Besides, nuvistor made of metal and ceramic is compatible with high vacuum. The mathematical model for Simulation Program with Integrated Circuit Emphasis (SPICE) of the nuvistor 8085 was created based on the experimentally measured static anode characteristics. The oscillator circuit was simulated by means of LTspice IV software for the circuit optimisation. The prototype of the oscillator printed circuit board was produced and tested. The results of the oscillator tests will be presented.

Id 808

Abstract Final Nr. P2.055

Laser cleaning of mirror surface for optical diagnostic systems of the International Thermonuclear Experimental Reactor

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Up-to-date tokamaks have a large complex of optical diagnostics for research of plasma parameters. Diagnostics located inside of the discharge chamber are exposed to intensive radiation effect, sputtering by charge-exchange atoms and contamination at the expense of redeposition of sputtered materials from different elements. Cleaning of mirrors by pulsed laser radiation is an effective method to restore the optical properties of mirrors [1, 2]. In the work an experimental study of the cleaning efficiency of carbon and metal-oxide films by an ytterbium fiber laser with wavelength 1.060-1.070 nm was carried out. The laser cleaning optimization of mirror surface has been carried out on the samples from polished metal substrates with hydrogenated carbon films on them. Such deposited films imitated chemical composition and conditions close to expected ITER conditions. The film thickness was 60-600 nm. It was shown that high initial reflection characteristics of optical elements can be recovered by choosing regimes of the effect of radiation on the surface with a deposited film. Efficient cleaning is ensured by radiation with the power density less than 107 W/cm2. At such power density the cleaning occurs in a solid phase and the thermal effect on the mirror is insignificant. Using the optimal regime it is possible to reduce the surface roughness of the mirrors and also to polish their surface. In this work is presented an overall optical scheme of the first mirror surface laser cleaning designed for system of active spectroscopy, located in the ITER equatorial port. Geometry and required characteristic of optical elements and devices for the system were defined. Main units of laser cleaning system of the first mirror surface were calculated and designed. [1] Buzhinskij O.I., et al., 20-th International Conference on Plasma Surface Interactions in Controlled Fusion Devices, 2012, P3-070 [2] Alexandrova A.S., et al., Journal of Physical Science and Application. 3 (1) (2013) 1

Id 147

Abstract Final Nr. P2.056

Conceptual Plant System Controller Architecture for ITER Magnetics Diagnostic

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The magnetics diagnostic will be one of the key plant systems required to guarantee effective and safe operation of the ITER tokamak. Besides providing important data for plasma physics studies, it will also have to supply reliable data for machine protection and for basic and advanced plasma control. The system includes a state-of-the-art data acquisition system capable of integrating in real-time, with extremely low drift variations, the voltages measured by a large set of distinct magnetic probes, with a plant architecture that has to guarantee very high reliability of the functions serving plasma control and machine protection. The latter requirement can conflict with the diagnostic operator’s need for sufficient freedom to update and tune the diagnostic parameters, allowing for the plasma physics output to be continually optimized as more knowledge and subtleties about the machine operation becomes available. This work presents a conceptual architecture for the magnetics diagnostic plant system controller that tries to address the aforementioned issues. It starts with an overview of the magnetics diagnostic and introduces the key user-requirements. A description of the foreseen data acquisition solution, focusing on the problem of the integration of the voltages from the inductive magnetic probes, is discussed together with the diagnostic measurement requirements. This is followed by an analysis of the reliability of the services for machine protection and basic plasma control applications and how these are addressed by an optimal distribution of the sensors in the data acquisition units. Finally, the interfaces to the ITER CODAC system are discussed, with the aim of anticipating missing features and misconceptions that could steer the design towards a sub-optimal solution in terms of allocation and delegation of resources and services. Finally, the roadmap from conceptual to final design of the plant system is presented.

Id 418

Abstract Final Nr. P2.057

An Improved Model for the oPtImal MeasurementProbes Allocation Tool

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The strategies and methodologies that are used to select the geographical distribution of the connections between measurements and data acquisition systems will have an impact in the reliability of the functions implemented by a given system. The oPtImal Measurement Probes Allocation (PIMPA) tool [1] is based on the solution of Integer Linear Programming (ILP) models and tries to maximize the reliability of a system function by proposing a signal connection layout that minimizes the effect of the failure of a data acquisition component. The model uses as input the dependencies of all the functions on the measurement signals, the relative priority of the functions (which is related to their importance), and the data acquisition parameters, which are defined in terms of number of input signals, number of slots in the chassis, and number of channels per slot. The first PIMPA model [1] was successfully assessed using as an use-case a simplified definition of the ITER magnetic diagnostic system. Nevertheless, this instance of the model had some important shortcomings, namely, it did not support the concept of individual slots (i.e. signal probes were only allocated to a given chassis) and, driven by the large number of probes (> 1000), the input signals had to be grouped in coherent sets (e.g. by type or by location). This work presents an improved ILP model that tries to address the above mentioned problems. It is shown that the model is capable of handling a more realistic definition of the ITER magnetic diagnostic system, taking into account all the individual probes and a more accurate definition of the magnetic diagnostic functions (e.g. plasma current, shape, velocity, MHD instabilities, etc.). Finally, a comparison between the results using the two models is discussed together with the possibility of using the tool in other diagnostic systems. [1] G. De Tommasi, A. C. Neto and C. Sterle, “PIMPA: a tool for oPtImal Measurement Probes Allocation”, accepted for publication in IEEE Transactions on Plasma Sciences

Id 418

Abstract Final Nr. P2.058

Simulation study on Doppler shift spectroscopy for neutral beam injection

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Doppler shift spectroscopy (DSS) is developed for decades as a mature tool to determine the key neutral beam parameters both for heating and diagnostic purpose. During the construction phase of diagnostic beam injection system for the Joint Texas Experimental Tokamak (J-TEXT), a set of simulation package, which is inspired by Manfred's work (Simulation of Spectra), is developed to guide the DSS installation. In the simulation work circular surface is adopted for the ion source with a number of beamlets to offer uniform or non-uniform intensity distribution. Energy distribution analysis is mainly focused on beam acceleration geometry and beam perpendicular temperature . Ratio of energy fractions is predefined and the multi-Guassian shifted Hα spectrum is reconstructed after neutralization and excitation procedure. Line broadening is contributed by beam parameters and collecting optics. This work is also expected to give DSS predictions between beam shots.

Id 80

Abstract Final Nr. P2.059

The information management within the Linear IFMIF Prototype Accelerator (LIPAc) commissioning

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The development of IFMIF (International Fusion Material Irradiation Facility) to generate a source of neutrons is indispensable to qualify suitable materials for the First Wall of the nuclear vessel in fusion power plants. As part of IFMIF validation activities, LIPAc (Linear IFMIF Prototype Accelerator) facility, currently under installation at Rokkasho (Japan), will accelerate a 125mA CW and 9MeV deuteron beam with a total beam power of 1.125MW. The commissioning of LIPAc has been divided into three major stages that will provide, given the state-of-the-art technologies that will be tested, a vast amount of complex data that will have to be processed and rapidly analyzed. In order to optimize the commissioning time and maximize the profit from the operational experience, a number of software tools could be used in the control room as part of future commission and operation activities. This paper will analyze the impact that information management tools, as well as other software tools specifically developed for the commissioning of LIPAc, might have in the overall commissioning phase. Tools such as adapted archiving system interfaces, electronic logbooks, post mortem analysis tools or even simulated test scenarios have already been proven of extreme utility in other similar scientific installations, where time intervals for decision-making between test scenarios had an enormous impact in the overall schedule. In addition, given the particularity of LIPAc as test bench for the future accelerators of IFMIF, the use of a performing knowledge database would make possible the transfer of the experiences acquired and implement them in IFMIF.

Id 466

Abstract Final Nr. P2.060

Design of data acquisition and control system for Indian test facility of diagnostics neutral beam

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The Indian Test Facility (INTF)-a negative Hydrogen ion based 100kV, 60A, 5Hz modulated NBI system having 3s ON/20s OFF duty cycle. Prime objective of the facility is to install a full-scale test bed for the qualification of all diagnostic neutral beam (DNB) parameters, prior to installation in ITER. The automated and safe operation of the system will require a reliable and rugged instrumentation and Control system which provide control, data acquisition (DAQ), interlock and safety functions, referred as INTF-DACS. The INTF-DACS has been decided to be design based on the ITER CODAC architecture and ITER-PCDH guidelines since the technical understanding of CODAC technology gained from this will later be helpful in development of plant system I&C for DNB. For complete operation of the INTF system, approximately 900 nos. signals are required to be superintending by the DACS. In INTF-DACS, conventional control loop time required is in the range of 5ms–100ms, therefore for the control system, PLC system (Siemens S-7 400 and 300) and for the DAQ except high end diagnostics, required sampling rates in range of 5 sample per second (Sps) to 10 kSps, therefore National Instruments (NI) PXIe system and NI 6259 digitizer cards; have been selected from the ITER slow and fast controller catalogs respectively. For high end diagnostics DAQ required sampling rates up to 100 MSps which is normally in case of certain events, therefore event based data acquisition hardware has been finalized. CODAC core software (CCS) for control application and NI-Labview for the DAQ application has been finalized due to full required DAQ support is not available in present version of CCS. Interlock and occupational safety system, for investment protection of facility and safety of the involved personal, respectively, have been design as per ITER-PCDH recommendations, which are based on IEC 61508 standard. This paper described the details design of INTF-DACS based on ITER-PCDH guidelines.

Id 229

Abstract Final Nr. P2.061

Diagnostics of plasma rotation using high-resolution spectroscopy on the COMPASS tokamak

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High-resolution spectroscopy is a powerful tool for measurement of both ion temperature as well as plasma rotation using observation of Doppler broadening of the emitted spectral lines and their Doppler shift, respectively. Both passive and active variants of this diagnostic for the COMPASS tokamak will be introduced. The passive diagnostic focused on the carbon triplet (C III at 464.74 nm, 465.02 nm and 465.15 nm) is utilized for observation of the plasma poloidal rotation on COMPASS. The set-up of the measuring system will be described, including the presently used imaging objective and fibre optics. The intended exchange of these two parts for a combination of the toric lens with the standard 50 mm objective coupled with a new fibre bundle leading to a significant enhancement of the diagnostic throughput will be discussed. Different options of increasing of the fibre collection area will be mentioned, including a flower-like fibre bundle, a use of micro-lenses, and a technology of the fibre narrowing. The contribution also shows the most recent measurements of poloidal plasma rotation of order of 0-6 km/s in the current set-up. The design of the new active diagnostic for COMPASS using a deuterium beam (heating neutral beam) and based on Charge Exchange Recombination Spectroscopy will be introduced. The tool will provide both space (better than 2 cm) and time (about 5 ms) resolved ion temperature and toroidal plasma rotation profiles. Measurements will use the C VI line at 529.05 nm ($n = 8 \rightarrow 7$) emitted from the charge exchange reaction of fully ionized carbon ions with beam-accelerated deuterium atoms. The results of the Simulation of Spectra code used to examine the feasibility of charge exchange measurements on the COMPASS tokamak will be shown and connected with a selection of the high throughput Czerny-Turner spectrometer coupled with the two-dimensional CCD camera.

Id 422

Abstract Final Nr. P2.062

Soft X-ray tomographic reconstruction of JET ILW plasmas with tungsten impurity and different spectral response of detectors

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At JET, the spatial distribution of soft X-ray radiation (SXR) is measured by horizontal (H) and two vertical (V, T) pinhole cameras. Although tomographic reconstruction of SXR emissivity from their data is challenging at JET due to different toroidal positions and spectral sensitivities of the cameras, robust and useful information can be obtained with optimised algorithms. With the ITER-like wall (ILW), interest in the SXR data analyses increased due to studies of tungsten transport. A forward-fitted deconvolution of SXR emissivity based on a coronal model of plasma demonstrated that different spectral sensitivity of H camera (350 micron Be filter) with respect to V and T cameras (250 micron Be filter) can cause significant inconsistency in the data due to spectral properties of tungsten radiation. While the inverse SXR tomography maintains the advantage of being independent of plasma parameters, it is necessary to correct any major discrepancy in vertical and horizontal response to tungsten radiation in order to avoid artefacts in the reconstructed image. In previous works, a trivial correction based on multiplying all V and T data by a constant was introduced so that the total plasma emissivity is equal in both vertical and horizontal directions. In the present contribution, possible artefacts due to the inconsistency of observed vertical and horizontal emissivities during tungsten events are studied, and an advanced correction based on adjustments of the constant for individual lines of view of the SXR detectors is presented. The adjustments take into account atomic data of tungsten as well as the SXR intensity perturbation due to tungsten influx. The results are evaluated in terms of goodness-of-fit and smoothness parameters corresponding to the reconstructed image. The applicability of this method to directly estimate tungsten density from the SXR data is discussed.

Id 781

Abstract Final Nr. P2.063

Investigation of metal Hall sensors for local magnetic field measurements at fusion reactors

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Hall sensors with their small dimensions, simple principle of operation, linear dependence of output voltage on measured magnetic field, 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. Specially developed semiconductor Hall sensor based on InSbxAs1-x show promissing results up to temperature of 300 degC and neutron fluences of the order of 10^18 cm-2. However, further hardenning of these semiconductor devices up to the radiation and thermal environment to be expected within the DEMO-like fusion reactor blanket structure is not straighforward. 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 supposed 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. We developed technology for manufacturing of Hall sensors with copper and bismuth sensing layer respecting constrains posed by harsh environment of fusion reactors. Characterisation of the sensors properties was done using AC detection technique to ensure high noise immunity. The proposed contribution will review the present optimized design, including some alternative technology options, and description of parameters of the resulting sensors: offset voltage, sensitivity and its dependence on temperature, input and output resistance, 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.

Id 829

Abstract Final Nr. P2.064

Li-BES detection system for plasma turbulence measurements on the COMPASS tokamak

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The new Li-BES diagnostic system with beam energy ranging from 10 keV up to 120 keV has been recently installed on the COMPASS tokamak. The diagnostic allows us to reconstruct density profile on the base of measured light profiles, and to follow turbulent behavior of the edge plasma. The technical requirements for fast Li-BES measurement on the COMPASS tokamak made it necessary to design and realize the state of the art Avalanche Photodiode Detector (APD) system. The 1:1 imaging relay optics has been designed including focusing cylindrical lenses in front of each detector panel. Each detector panel consists of the APD detector plate integrated with temperature stabilized broadband amplifiers. Analogue signals from the APD detectors are transmitted to a 32-channel ADC with 2 MHz sampling rate. The field of view of the APD system covers the entire minor radius at the outer midplane of the COMPASS tokamak. The spatial resolution of the measurement system is about 1-2 cm depending on the beam energy. In order to obtain reliable experimental data, a calibration of the system has been performed. The calibration consists of the spatio-temporal calibration and the calibration of relative sensitivity of the individual detectors. As a demonstration of the working principles, detailed fluctuation measurements have been performed in various plasma configurations. Both spatio-temporal properties of turbulent fluctuations as well as their statistical description as a function of radial coordinate have been deduced. In this paper we present the diagnostic for turbulence studies - the array of 18 APDs combined with appropriate optics and interference filter. Also, the results of its calibrations and results of spatio-temporal characterization of COMPASS plasmas using the new Li-BES system will be presented.

Id 928

Abstract Final Nr. P2.065

Validation of equilibrium tools on the COMPASS tokamak

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Various MHD (magnetohydrodynamic) equilibrium tools, some of which being recently developed or considerably updated, are used on the medium-size COMPASS tokamak [R. Pánek et al., Czech J Phys 56, B125, 2006]. MHD equilibrium is a fundamental property of the tokamak plasma, whose knowledge is required for many diagnostics and modelling tools. Proper benchmarking and validation of equilibrium tools is thus key for interpreting and planning tokamak experiments. We present here benchmarks and comparisons to experimental data of the EFIT++ reconstruction code [L.C. Appel et al., to be submitted to Nucl. Fusion], the free-boundary equilibrium code FREEBIE [J.-F. Artaud, S.H. Kim, EPS 2012, P4.023], and a rapid plasma boundary reconstruction code VacTH [B. Faugeras et al., PPCF 2014, accepted]. We demonstrate that FREEBIE can calculate the equilibrium and corresponding poloidal field (PF) coils currents for given plasma parameters. Both EFIT++ and VacTH can reconstruct equilibria generated by FREEBIE from synthetic diagnostic data (including data with artificial errors) as well as from actual experimental data. Optimum reconstruction parameters are estimated for both codes; in addition, possible enhancements using more diagnostics are discussed and simulated using synthetic diagnostics. FREEBIE can also calculate the temporal evolution of the poloidal field coils currents for a whole plasma scenario. Possibilities for employing VacTH in a real-time control of the plasma shape are discussed.

Id 963

Abstract Final Nr. P2.066

Millimetre wave attenuation of prototype diagnostic components for the ITER bolometers

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The bolometers in ITER are vulnerable to stray radiation from electron cyclotron resonance heating (ECRH) which results in measurement errors for plasma radiation detection. To protect the detectors from this stray radiation in the millimetre wavelength range, dedicated diagnostic components have been designed and tested. One option is to place a top plate which contains a microwave-reflecting grid onto the collimators. Another option investigated is the coating of the collimator channels using a mixture of Al2O3 and TiO2, a microwave absorbing ceramics. Measurements of the millimetre wave attenuation of the collimator in front of the bolometer detectors with and without top plate or coated collimator channels have been performed in the frequency range of 125 – 420 GHz. The attenuation factor of the collimator channels including the top plate is at least 10^7 at 170 GHz (the ECRH frequency for ITER), which is sufficient to reduce the residual ECRH induced signal significantly below the one due to plasma radiation. The collimator channels without the reflecting grid have a relatively low attenuation factor of 10, typically. However, coating the channels with microwave absorbing ceramics increases the attenuation to at least a factor of 10^4. Transmission measurements of the complete bolometer head with isotropic radiation confirm these results within the available dynamic range. Additionally, measurements have been performed in the microwave test facility MISTRAL, which provides an isotropic microwave field of 750kW/m^2 at 140GHz within a large volume. Placing a complete bolometer camera inside MISTRAL, the attenuation factor of the full diagnostic set-up using a top plate was determined to be in the order of 5•10^4. This implies that particular attention has to be paid to design and quality control of the joints of diagnostic components to prevent microwave leakage.

Id 344

Abstract Final Nr. P2.067

Integrated thermal fe analyses and testing of prototype components for the iter bolometer diagnostic

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The thermal design of diagnostic components for the ITER bolometer diagnostic is of critical importance with respect to survivability, reliability and performance. To support the development of bolometer camera prototypes for ITER at IPP, KRP-M has performed the thermal analysis to determine the temperature distribution, to identify critical items and uncertainties, and to optimize the camera design to achieve low detector temperatures. The analysis has been carried out in close interaction with tests for reducing the uncertainties of those material parameters responsible for the highest uncertainties in the results and for verifying the simulation model. The FE model is introduced in detail and key results are presented. One major outcome of the design optimization has been to modify the geometry in such a way that the heat flow imposed by outer radiative heat loads is bypassed around the detector area to achieve low detector temperatures. The material parameters identified as most critical and featuring considerable uncertainties in model assumptions are the emissivity for the radiation to the ambient as well as the thermal contact conductivity of the various bolted connections of the camera. Specific tests have been designed and performed to determine these quantities experimentally. Emissivity values for TZM have been determined depending on surface quality ranging from 0.05 to 0.38. For the thermal contact conductivity, a multi-level test has been performed to investigate its dependency on various design parameters as e.g. contact pressure. For TZM/TZM contact for example, values of 1600 to 20000 W/m2K have been determined for contact pressures from 1 to 30 MPa and typical surface finishing. For the verification of the complete FE model, a thermal balance test with a prototype has been designed and performed. The results show a very good agreement between model and experiment with respect to temperature distribution.

Id 944

Abstract Final Nr. P2.068

Impact of error fields on equilibrium configurations in ITER

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Error fields in ITER are considered one of the major sources of possible loss of performance in the actual device with respect to the design. As a matter of fact, small discrepancies of magnetic field from the nominal one will interact with plasma evolution leading to birth of magnetic islands and finally loss of stability. This issue is well known in present experimental Tokamaks, and it is expected to be even more relevant in larger machines such as ITER. The error field level driving plasma to loss of stability can be estimated using numerical models and/or available experimental data, giving rise to estimates of “acceptable” error fields. Error fields in actual devices are then counteracted using suited correction coils, to achieve satisfacory field maps. On the other hand, the effect of the error fields on plasma equilibrium configuration is still being actively investigated. In this work, the analysis of the effect of residual error fields on the ITER axisymmetric plasma configurations wll be performed. First, a number of deformed coils configurations are considered, and the corresponding error fields are computed by means of the MISTIC code [1]. Then, the CarMa0NL code [2], able to solve nonlinear Grad-Shafranov equations in presence of 3D conductors, is used to evaluate the subsequent perturbation of the plasma axisymmetric equilibrium configuration. Various cases will be presented, in order to highlight the expected effect of specific coil deformations. Moreover, also a “statistical” overview of possible consequences of specific sets of deformations will be carried out. This work was supported in part by Italian MIUR under PRIN grant 2010SPS9B3. [1] A. G. Chiariello. A. Formisano, R. Martone, Fus. Eng. Des. 88 (2012), DOI 10.1016/j.fusengdes.2013.02.124 [2] F. Villone et al., Plasma Phys. Control. Fusion 55 (2013) 095008

Id 722

Abstract Final Nr. P2.069

Reference design of the Power Supply system for the Resistive-Wall-Mode control in JT-60SA

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JT-60SA is the satellite tokamak under construction in Naka, Japan, in the framework of the EU–JA “Broader Approach” Agreement. In JT-60SA, to attain steady-state high-beta plasmas, suppression of Resistive Wall Modes (RWM) is necessary. At this purpose, a passive stabilizing plate (SP) and an active control system based on 18 in-vessel sector coils (SC) are foreseen. In the past years, physics studies have been carried out to produce the inputs for the design of the RWM control system and the SC design evolved accordingly. In the present arrangement, the coils are placed on the plasma side of SP, to minimize the magnetic shielding effects of passive structures. Electromagnetic analyses on SC have been carried out in support to the evaluation of the different coil design solutions and to derive the requirements for the related power supply system (RWM-PS). To achieve the highest flexibility in RWM control, it has been devised to feed each SC independently, with currents following 18 real-time references. In 2012, the SC design has been further improved by increasing the number of turns from 2 to 8. This allowed reducing the rated current, thus the voltage drops on the feeders and the power required from RWM-PS. To keep the RWM under control with relatively low Ampere-turns (2.2 kAt), the RWM-PS has to guarantee very fast dynamics (current bandwidth: 3 kHz; maximum latency: 50 us), largely beyond that of standard industrial applications. This paper firstly reports the main requirements for the RWM control system. Then, the reference design of the RWM-PS is described. It includes an ac/dc conversion system, dc-link capacitor banks and a set of 18 fast inverters. The advantages of the proposed scheme are discussed and the main electrical parameters are shown in detail. The main requirements of the control section are given, with details on possible implementation and interfaces with JT-60SA central control.

Id 612

Abstract Final Nr. P2.070

Improvement of the dynamic response of the ITER Reactive Power Compensation system

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The ac/dc conversion system necessary to supply the superconducting coils of the magnet system and the auxiliary heating systems of ITER experiment can absorb a total active and reactive power up to 500 MW and 950 Mvar, respectively. The allowable limit value of reactive power absorbed from the grid is 200 Mvar; therefore a Reactive Power Compensation system (RPC) rated for a nominal power of 750 Mvar will be provided. The current reference design of the) is based on Static Var Compensation technology with Thyristor Controlled Reactor (TCR) and Tuned Filter. The RPC has to minimize the demand of reactive power from the grid; its control is based on a feed-forward method, where the corrective input is the measurement of the reactive power consumption of the ac/dc converters. This is derived from the Fast Fourier Transform (FFT) at 50 Hz of the measurements of the three-phase voltages and currents, then this signal is converted in a firing angle of the TCR thyristors by a lookup table. The delay introduced by the FFT calculation and the slow response of the TCR could make the response speed of the RPC not sufficient to face the fast variations of the reactive power demand, causing the fast voltage variation of the 66 kV busbar, and this can occur as a consequence of the Plasma Control System (PCS) operation, for example vertical displacement control, or intervention of the fast discharge units. In this paper a new controller of the RPC able to overcome this shortcoming is proposed. It is based on the calculation of the predicted consumption of the reactive power by using the voltage reference signals coming from the PCS and the dc current measurements of the ac/dc converters, and on the speed up of the RPC control by introducing a lead-lag transfer function. Its effectiveness is verified in different operating conditions of the ac/dc conversion system.

Id 616

Abstract Final Nr. P2.071

Improvements of the RFX-mod power supply system for MHD mode control

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RFX-mod is a Reversed Field Pinch (RFP) machine for plasma confinement operating at up to 2MA plasma current. Key feature of RFX-mod is the large number of saddle coils (192) that completely cover the outer surface of the stabilising shell and allow the generation of precise local magnetic field to control the plasma instabilities. Each coil is fed by a dedicated switching inverter with onboard full-digital control. For more than one decade since its installation, this power supply (PS) system has being showing a very effective and versatile control of MHD modes in many RFX-mod plasma scenarios: mitigation of Tearing Modes, control of spontaneous RFP helical states and Resistive Wall Modes. A central MHD mode control system generates in real-time the 192 reference signals to drive the inverters. Recently a new faster central system based on the MARTe real-time package has been implemented. To take advantage of the effectiveness of the new control, an improvement of the PS dynamics is devised. To the purpose, a review of the inverter control has been undertaken with the main objective of reducing the latency. Furthermore, the possibility to operate the inverters in open loop mode has been studied, to achieve a faster response of the overall control system of the MHD modes. The paper will report the analyses and preliminary experimental results for both solutions.

Id 713

Abstract Final Nr. P2.072

Analyses of the impact of connections layout on the coil transient voltage at the Quench Protection Circuit intervention in JT-60SA

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The interruption of high direct currents, required in fusion experiments both for protection and energy transfer systems, is a challenging task to be managed safely for the whole plant. Depending on the technology of the circuit breakers the resulting current derivative may be high, leading to dangerous transient overvoltages causing stress on insulations of the connected devices. This aspect has been analyzed for JT-60SA Quench Protection Circuits (QPC), the system devoted to the protection of poloidal (PF) and toroidal (TF) field superconducting coils. The QPC adopts edge technology solutions for current interruption: a Hybrid mechanical-static Circuit Breaker (HCB) as main circuit breaker in series with a PyroBreaker (PB) as backup protection; both impose high current derivative and relevant fast transient voltages when operated. Snubbers or clamp networks can be provided in parallel to the breakers to smooth the voltage waveform; their effectiveness depends not only on their design but also on their location in the circuit and on the stray impedance of their connections. Dedicated clamp networks for the HCB of PF and TF QPC have been designed and tested during the qualification of the QPC prototypes. On the contrary, it was preferred not to apply any component in parallel to the PB, the ultimate protection of the coils, to avoid reducing its reliability. For PB a different approach has been worked out, based on the optimization of the layout of the QPC connections. Analyses have been performed to reproduce the transient voltage across the TF coils at the PB intervention so as to highlight the impact of different busbar routes on the surge voltage. The results indeed showed a variation of the peak voltage in between ± 30% of the maximum allowed value. The reliability of the results of such analyses is strongly affected by the correctness of the QPC model, which has been validated against experimental results of the test campaigns on QPC prototypes and by the accuracy of the estimations of the busbars' stray impedances as in the final installation arrangement. The paper will present the analyses carried out and will discuss the results.

Id 742

Abstract Final Nr. P2.073

Twin box ITER joints under electromagnetic transient loads

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The ITER Toroidal Field (TF) Coil winding packs are designed to be wound in double pancakes using the 68 kA Nb3Sn conductors. The twin box joint concept provides the electrical pancake-to-pancake joints between the two Nb3Sn conductors and coil-to-bus bar joints between the TF coil 68 kA Nb3Sn and bus bar 68 kA NbTi conductors. The twin box full size joint sample connecting the two Nb3Sn conductors was prepared in order to investigate the TF joint design in SULTAN Test Facility. The original goal of the test campaign was to measure the resistance and AC loss at different operating conditions. The accidental dump of background field caused the noticeable increase of resistance due to induced electromagnetic transient load at the end of test campaign. The TF joint test was continued in January 2014 in order to investigate the TF twin box joint design under the periodical electromagnetic transient load by triggering the dump of background field intentionally. The test results of TF twin box joint under the periodical electromagnetic transient load are presented and the 68 kA Nb3Sn conductor performance observed during those tests is highlighted in this paper.

Id 314

Abstract Final Nr. P2.074

Soldered Lap Joints between REBCO Coated Conductors for Demountable Fusion Magnets

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Recent advances in the high temperature superconductor (HTS) technologies improve the possibility of including demountable HTS magnets in commercial nuclear fusion reactors. A thorough understanding of different types of joints is essential to optimise their performance. In this work we investigate properties of lap joints consisting of two Rare-Earth-Ba2Cu3O7 (REBCO) coated conductors soldered together. We report numerical modelling data, critical current density (Jc) and resistivity measurements on the jointed conductors from 300 K to 4.2 K in applied magnetic fields up to 12 T and scanning electron microscopy (SEM) studies. Numerical analysis shows that the resistive current density is almost uniform within the solder. Although the temperature at the centre of the joint can be large, adding low resistivity copper layers to the joint can reduce the maximum temperature in the joint and improve its stability. Resistivity measurements and SEM data indicate that in the superconducting state, the joint resistivity predominately has two sources, the solder layer (~20 - 40 nOhms-cm2) and the interfacial resistivity of the REBCO/silver interface within the coated conductors (~20 nOhms-cm2). Our data show that the magnetic field dependence of the interfacial joint resistivity is highly anisotropic and the anisotropy increases as the temperature decreases. Jc measurements confirm these resistivity findings for currents up to 300 A. To optimise demountable joints for fusion machines will require consideration of the positioning of joints because of the large anisotropy of the magnetoresistance of the component interfacial resistivities, use of potentially superconducting solder, management of heat extraction from the joint, and careful materials choices for the components of the joints to enable straightforward remote assembly. Acknowledgements: We would like to thank: Simone Smith1, Prapaiwan Sunwong1, Mark J. Raine1, Robert Edge1 and Tom Lee1,2. This work was funded by the RCUK Energy Programme under grant EP/I501045.

Id 426

Abstract Final Nr. P2.075

Numerical study for optimization of the air cooling system for the Fast Discharge Resistors protecting the ITER magnets

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The superconducting magnets of the present-day fusion installations, when in operation, accumulate huge energy. Thus, for example, ITER magnets are capable of accumulating up to 50GJ. In case of coil quench or a serious failure of the power supply system the energy stored in the coils must be extracted rapidly. The problem can be solved by the Fast Discharge Resistors (FDR) under development at the Efremov Institute, which would dissipate the energy discharged during coils “warm-up”, thus protecting the magnets against failure. The FDR system is made as a set of sections consisting of a resistive element enclosed in a steel casing. Two-four vertically stacked sections form a separate module. The fast discharge of the coils results in practically adiabatic heating of the resistive elements up to 300-350°C. The target was to cool the resistors to their initial temperature within 4-6 hours. With this in mind, the authors have designed the air cooling system based on air natural circulation developed in the system of channels formed by the supply/return piping, vertical modules and chimney. The air natural circulation allows one to save equipment, space and energy necessary for air pumping. When performing the numerical analysis of the cooling process, the authors faced the problem of the essential non-uniformity of air flow distribution in parallel vertical modules, which increases considerably the cooling time. Thus, if for a single FDR module the cooling time does not exceed 2-3 hours, the cooling time for several modules exceeded already 20 hours. The numerical studies performed over the last few years allowed the authors to propose a number of measures on optimizing the air cooling system so as to mitigate the negative effect of the air flow non-uniformity in the FDR module system and, finally, to reduce significantly the cooling time.

Id 677

Abstract Final Nr. P2.076

Conceptual Design of Pulsed High Voltage and High Precision Power Supply for Plasma Heating by a Cyclotron Auto-Resonance Maser (CARM)

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Due to the high electrons' temperature during the plasma burning, both a higher power (>1 MW) and a higher frequency (up to 300 GHz) are required for plasma heating in future fusion experiments like DEMO or FAST. For this task, ENEA started a project to develop a Cyclotron Auto-Resonance Maser (CARM) able to produce an electron radiation in synchronism with the RF field and to transfer the electron beam kinetic energy to the plasma. This facility requires an advanced pulsed power supply with the following technical specifications: • Variable output voltage up to 700 kV; • Variable pulse length in the range 5-100 microseconds; • Overshoot <2%; • Rise time <1 microseconds; • Voltage accuracy <0.1%, including drop, ripple and stability. The studies on such system and the achieved results are summarized in the following. The preliminary design consists of four main parts: 1. An AC/DC converter with transformer. 2. A IGBT-based DC/DC pulse modulator with output voltage of 4 kV. 3. A multi-primary pulse transformer with 8 primary windings and one 700 kV secondary winding. The pulse transformer parameters were determined according to scientific literature and IEEE standards, considering a pulse width up to 100 microseconds. 4. A pulse width modulation (PWM) PID-based control system, optimized to follow very fast rectangular pulses. The control system starts the modulation when the voltage on the DC/DC modulator capacitance reaches a fixed threshold. A MATLAB/Simulink model was set up in order to identify the best layout, configuration and parameters for the system, in particular for the pulse transformer and modulator. The performed simulations showed that the designed power supply is able to satisfy the technical specifications. Moreover, the modulator configuration allows a high flexibility in terms of pulse magnitude and duration, fast dynamic response and high precision and the controller highlights a high stability.

Id 115

Abstract Final Nr. P2.077

Design and realization of JT-60SA Fast Plasma Position Control Power Supplies

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JT-60SA is a Superconducting Tokamak in the framework of the Broader Approach Agreement (BAA) between Europe and Japan, a program of complementary facilities to be realized in parallel to ITER. Under this BAA, the Italian National Agency for New Technology, Energy and Sustainable Economic Development (ENEA) has to provide eight PSs and six transformers for the JT-60SA PSs, including the Central Solenoid (CS1, CS2, CS3, CS4), Equilibrium Field (EF1 and EF6) PSs and the Fast Plasma Position Control (FPPC) PSs Upper and Lower. These systems are being procured by ENEA through a contract signed in August 2013 with Industrial Suppliers Poseico and Jema in Temporary Association of Enterprise. The installed total power is about 400 MVA, and a reactive power's peak of 330 MVAR. The basic devices of a FPPC PS consist in: a FPPC coil, a thyristors rectifier or base PS, a converter transformer, a crowbar (to protect by over-voltages). The first FPPC PS will be tested by September 2014. FPPC PSs control vertical position of the plasma during a plasma shot, against Vertical Displacements Event (VDE). For this task, the FPPC PSs have to be as fast as possible and provide as much voltage as possible. Further, these will be open loop feed forward voltage controlled. The main characteristics are: 4-quadrant AC/DC converter 12-pulse with circulating current, DC load voltage ±1000 V and DC load current ±5 kA. The induced overvoltage in FPPC coil during a plasma disruption can reach 10 kV and it is bounded by a nonlinear resistor in parallel to the crowbar. All these technical characteristics have strongly influenced the design of the FPPC converter and of the FPPC transformers, that have been validated by simulation model of FPPC PS. The outcomes of the simulation have allowed to finalize the performances and dynamic behavior of voltage response.

Id 115

Abstract Final Nr. P2.078

First Switching Network Unit for the JT-60SA superconducting Central Solenoid

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The Central Solenoid (CS) of the new international tokamak JT-60SA is divided in four superconducting modules. Each module is connected to an independent power supply circuit including a 4-quadrant AC/DC converter and a Switching Network Unit (SNU). The main scope of the SNU consists in inserting proper resistors in the circuit in order to generate and sustain the loop voltage needed for plasma breakdown and initiation. In the framework of the “Broader Approach” agreement between Europe and Japan, the four CS SNUs will be procured by the Italian Agency ENEA. The first SNU (full-scale and fully-working prototype), consisting of six cubicles, was assembled in 2013. The factory tests on this system, including tests at full current and voltage, are being performed throughout 2014. The manufacturing of the remaining three SNUs will proceed after the success of such tests. The main characteristics of the developed SNU are: • Synchronized use of an electronic static circuit breaker (SCB) in parallel with an electromechanical bypass switch (BPS) to exploit the benefits of both devices. • Nominal voltage of 5 kV, with a specific circuit to limit the transient voltage to 5.5 kV in order to preserve electronic and superconducting components.
• DC current interruption up to 20 kA (with a safety margin up to 25 kA). • Light BPS with opening and closing times shorter than 15 ms and 65 ms, respectively. • SNU effective opening/closing time shorter than 150 μs in nominal conditions (750 μs in worst case).
• Breakdown resistance adaptable from 0.25 Ω to 3.75 Ω by four selectors based on the desired scenario. • Fully electronic making switch (MS), able to reduce the SNU resistance up to 22 m Ω after successful breakdown to support plasma ramp-up. • Breakdown resistors that could dissipate much more than 90 MJ, with a maximum resistance variation of 8% at rated energy.

Id 828

Abstract Final Nr. P2.079

Validation of special processes for the manufacturing of the first JT-60SA TF coil

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In the framework of the Broader Approach program, ENEA awarded, in 2011, ASG Superconductors, in Genoa, Italy, a contract for the manufacturing of 9 TF superconducting coils of JT-60SA tokamak. TF coil manufacturing started in ASG after the production readiness review (PRR) for the winding operation in November 2013. This PRR was successfully achieved after the completion of several intermediate steps consisting in qualification of special processes. The qualification program has, however, not been concluded with the PRR because, after winding, other important manufacturing steps have to be fulfilled. Winding pack (WP) impregnation, WP insertion into coil casing components, casing welding and coil embedding are to mention, some of the most critical manufacturing phases that will be proved on actual coil size for the first time in the near future. Nonetheless, each of the mentioned processes has been already qualified through dedicated mock-ups. The validation program has now been completed and this paper sheds light on the most recent ones. In particular, welding of the casing components has been retained particularly critical for at least three reasons: i) during welding the WP may be damaged by the intense heating; ii) distortion caused by heating may determine incorrect coil geometry and then magnetic field errors; iii) flaws may reduce structural strength and then the overall lifetime of the machine. To investigate each of these aspects the qualification of the welding process has been divided in two steps. First the design of the weld seam geometry and of the weld process has been studied on plates with adequate thickness, then the process has been verified on 1-m-long casing mock-up with a WP mock-up in it. Similarly, final embedding has been demonstrated on another 1-m-long mock-up of the coil. Main results and lessons learned are here described.

Id 883

Abstract Final Nr. P2.080

The design and achievement of the double-fed system for the 100MVA motor generator of J-TEXT

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The 100MVA motor generator is the main power of J-TEXT, including the toroidal coil, the ohmic heat coil and the divertor coil and High-voltage power supply for Electron Cyclotron Resonance Heating(ECRH). The motor generator(MG) was designed in USA with a 60Hz grid so that the maximum stored energy was 185MJ. Now in China the grid is 50Hz ,so the maximum stored energy is 128MJ. To meet the design parameters of the MG, a doubled-fed control system with AC-DC-AC topology was designed to ensure the Tokamak operates with designed parameters. Using a rotational reference frame oriented by stator flux, speed and magnetic flux linkage can be controlled separately. And at the rotor side , the power factor of the converter can be 1. A prototype was built to verify the control system and a 250KVA converter was built to achieve the system.

Id 222

Abstract Final Nr. P2.081

R&D on High-Power DC Reactor Prototype for ITER Poloidal Field Converter

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This paper mainly introduces the research and development (R&D) of the high-power DC reactor prototype, whose functions are to limit the circulating current and ripple current in the ITER poloidal field (PF) converter. It needs to operate at rated large direct current 27.5 kA and withstand peak fault current up to 175 kA. Therefore, in order to meet the special requirements of the dynamic and thermal stability, a new prototype design structure of dry-type air-core water-cooling reactor with epoxy resin casting technique is presented, which is based on the theoretical analysis, finite-element simulation calculation and small prototype test verification. Now the full prototype has been fabricated by China industry, and the dynamic and thermal stability tests of the prototype have also been accomplished successfully. The test results are in compliance with the design and it shows the availability and feasibility of the proposed design, which may be a reference for relevant applications.

Id 219

Abstract Final Nr. P2.082

Tungsten coating for fusion plasma facing component development by thermal plasma spraying method

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Plasma facing materials for nuclear fusion reactor are very important in fusion performance and reliability. In order to develop plasma facing materials, many materials such as tungsten, beryllium, and carbon fiber components have been developed. Among them, the tungsten is recently favored because of high melting point and less sputtering effect in the plasma reactor. However, the tungsten is not easy to fabrication, needs thick block size to prevent thermal loss, and required high cost joining method to the different substrate. In this work, therefore, we studied thermal plasma spraying coating method for thin tungsten coating on the ferritic-martenitic steel. The plasma for coating is generated in 60 kW RF inductively coupled plasma source and 55 Kw DC vacuum plasma spray at low pressure around 100-350 Torr. Coating thickness was from 1 to 3 mm. The hardness investigated by Vicker hardness test was around 250 Hv. In order to characterize the coated tungsten layer, the surface morphology, micro-structure was investigated using SEM and XRD. The optimized condition for tungsten coated surface was studied by chaning powder feeding rate, input power, coating distance, plasma gas flow rates, sample pre-heat and post-heat condition. The heat load tes was performed using electron beam facility with 0.5 and 1.0 MW/m2 for tungsten coated sample and tungsten plate joined sample as a reference.

Id 488

Abstract Final Nr. P2.083

Characterization of Arc-Heated Plasma for Study of Plasma Material Interaction in Fusion Reactor Conditions

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Arc-heated plasma wind tunnel with 0.4 MW of power has been constructed at Chonbuk National University (CBNU) in Korea. Characteristics of plasma were investigted to study plasma material interaction in fusion reactor conditions. A segmented arc plasma torch was adopted as a plasma source, and it was designed to have high thermal efficiency and long life in the production of supersonic plasma flow with enthalpy above 13 MJ/kg at a velocity of Mach 3. This facility can provide both steady-state heat fluxes greater than 10 MW/m2 and ion fluxes greater than 1024 m-2 s-1 which are required to test plasma-facing components in a fusion reactor conditions. With flow rate of the argon in the range of 10 ~ 16 g /s and the input current in the range of 200 ~ 350 A, conditions to produce the required plasma beam are studied. In this paper the results of extensive diagnostics measurements for enthalpy, plasma densities and plasma temperatures using enthalpy probe, heat flux probe, emission spectroscopy, laser thomson scattering and various visualization systems are presented. It is shown that this facility can produce an ion flux above ~ 1024 m-2 s-1 as well as high heat flux above 10 MW/m2 relevant for the study of plasma material interaction in fusion reactor conditions.

Id 454

Abstract Final Nr. P2.084

HRP facility for fabrication of ITER vertical target divertor full scale plasma facing units

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ENEA and Ansaldo Nucleare S.p.A. (ANN) have being deeply involved in the European International Thermonuclear Experimental Reactor (ITER) development activities for the manufacturing of the inner vertical target plasma-facing components of the ITER divertor. During normal operation the heat flux deposited on the bottom segment of divertor is 5-10 MW/m2 but the capability to remove up to 20 MW/m2 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, named Hot Radial Pressing (HRP), suitable for the construction of these components. The last challenge is now to fabricate a full scale prototype of the IVT for the final qualification. On the basis of the experience of manufacturing hundreds of small mock-ups, ENEA designed and built a new suitable HRP facility. The tolerances of the plasma facing units (PFU) to be installed onto the supporting structure are fixed by ITER/F4E and are very tight. The objective of getting a final shaped PFU that satisfies these requirements is an ambitious target. The setting-up of the equipment started with the fabrication of full scale and representative 'dummies' in which stainless steel instead of CFC or W was used for monoblocks manufacturing. The results confirmed that the 'dummy'-PFU dimensions are compliant with the required tolerances. The paper reports a description of the innovative HRP equipment together with its characteristics and resolved issues. The dimensional check after HRP is also discussed. On the base of these results ENEA-ANN are now involved F4E-OPE-138 contract where the fabrication of full scale IVT units is requested.

Id 347

Abstract Final Nr. P2.085

Study of dynamic amplification factor of DEMO blanket caused by a gap at the supporting key.

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The work belongs to the design activities of the in-vessel components for DEMO that is the nuclear fusion power plant that will be build upon the expected success of the ITER machine. Among the several design aspects promoted by EFDA organization, namely materials, load specifications, remote handling etc, this work is related to system design analysis. The blanket segments requires a gap at all supporting keys. Due to its higher operating temperatures compared to the vessel ones, this gap will increase during operation. A movable element must be foreseen to close this gap during operation. The EM loads due to fast disruptions occur on a short time and might accelerate the blanket significantly before it touches the supporting keys, causing an impact of the blanket itself onto the keys. Depending on the stiffness of the blanket and its supports, the EM loads with their short time scale could excite the structure's natural frequencies, causing dynamic amplification. Both phenomena (impact and dynamic amplification) can cause stresses in the structure significantly higher than in the static stress state. The work deals with the development of a Finite Element Model of DEMO blanket suitable to study its nolinear transient dynamic behaviour under impact loadings. Moreover a sector of the vacuum vessel, the ribs between the inner and outer vessel, the backward manifolds and the supporting keys of the blanket have been modelled. Transient no-linear dynamic analysis have been performed with Abaqus and Ansys FEM codes mainly focused on the forces/displacement of the keys in their housing on the blanket. The dynamic amplification factor have been evaluated as the ratio of dynamic to static displacements/forces in meaningful points of the structure for a growing gap till to 5 mm. Also a study about the strain and kinetic energy has been carried on.

Id 528

Abstract Final Nr. P2.086

Design Optimization of the DEMO ITER-like water-cooled divertor

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A near-term water-cooled target solution has to be evaluated together with the required technologies and its power exhaust limit under 'DEMO conditions. The ITER-like design concept based on the mono-block technology using W as armour material and the CuCrZr as structural material with an interlayer of pure Copper represents the most promising solution. This work reports the design study of an "optimized" ITER-like Water Cooled Divertor able to withstand an heat flux of 10MW/m2 , in order to assess this design concept at DEMO relevant conditions. The optimization was performed by means of multi-variable optimization algorithms, considering the elastic behaviour of the materials by varying some geometrical parameters checking the thermo-mechanical behaviour of the mock-up (i.e. the max temperature, the max Von Mises stresses and the "elastic strain" in the CuCrZr), while the hydraulic conditions (inlet pressure, temperature and velocity) were fixed. Both for optimization study that for optimized mock-up a 3D model using 15 monoblocks of W armour having a thickness of 4mm with a gap of 0,25mm between each monoblock was used, an interlayer of Cu-OFHC was considered to soften the thermal stresses due to the mismatch for the thermal expansion coefficients between W and CuCrZr pipe. Additionally a swirl tape insert (0,8mm thick, twist ratio of 2) was modelled to promote turbulence inside the pipe. The hydraulic conditions were fixed to allow the CuCrZr pipe working in the suggested window operating temperature (200°-350°C). The optimized geometry was finally evaluated on the basis of the SDC-IC criteria and for the low cycle fatigue. Also the margin to the Critical Heat Flux was estimated because in the next future mock-ups will be fabricated and HHF tested to evaluate their performances.

Id 657

Abstract Final Nr. P2.087

Study of tile size effect on the thermal-mechanical performance of ITER beryllium first wall mock-up

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ITER FW panel consists of beryllium in the form of tiles covering its surface, high strength copper alloy as heat sink material, and stainless steel as structural material. Due to the thermal stress generated between beryllium and the heat sink material when the component is facing high energy deposition, the size of the beryllium tiles plays a critical role on the performance of the component. This paper presents the analysis of the tile size effect on the performance of the FW component. To this aim, a mock •]up (MU) supplied by Fusion for Energy was tested in the electron beam facility JUDITH2 at Forschungszentrum Juelich. The MU contains four beryllium tiles with two different dimensions, which are 141.8 mm .50 mm.10 mm and 94.2 mm .50 mm.10 mm, respectively. The MU was loaded cyclically under various surface heat fluxes (1 •]2.25 MW/m2) with ITER relevant coolant water conditions: inlet temperature 70 □‹C, pressure 3 MPa and velocity 3 m/s. After a total of 1800 thermal cycles, no visible damage was observed in the MU. In order to qualitify the temperature and stress distribution of the MU during the testing, a 3D FEM thermo •]mechanical analysis was performed using ANSYS14 workbench. The FEM calculated temperature is in good agreement with the experimental measured temperature. The FEM analysis results show that, in this particular case, the size of the beryllium tiles has a limited effect on the surface temperature of beryllium while the beryllium tiles with smaller size exhibit less stress concentration than the ones with bigger size.

Id 241

Abstract Final Nr. P2.088

Characterization of ITER tungsten and CFC qualification mock-ups exposed to high cyclic thermal loads

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As part of the tungsten divertor qualification program high heat flux (HHF) tests were performed in the electron beam facilities FE200 at AREVA Le Creusot, France and ITER Divertor Test Facility at Efremov Inst. St. Petersburg, Russian Federation. Thereby, in total more than 20 small-scale tungsten monoblock mock-ups manufactured by Plansee and Ansaldo, as well as Carbon Fiber Composite (CFC) monoblock mock-ups manufactured by Plansee were exposed to 10 seconds cyclic (considered as steady state) heat loads. The applied power density ranges from 10 MW/m2 (5000 cycles) to 20 MW/m2 (up to 1000 cycles). All the mockup withstood the required thermal cyclic loads 10 MW/m2 (5000 cycles) to 20 MW/m2 (300 cycles) and even higher cycle number at 20 MW/m2. This demonstrated the availability of tungsten divertor monoblock technology in the EU industry. Post-test investigations showed macroscopic cracks on some monoblocks, which developed almost symmetrically along the cooling tube axis, so-called, self-castellation. This self-castellation was parallel to the heat flux and did not impair the heat transfer from the loaded surface to the cooling tube. It was found that this crack was initiated at the loaded surface and propagated in some case down to the joint to the soft Cu layer but with no impact on the integrity of the cooling tube. Destructive examinations showed an extent of tungsten recrystallization of the heated surface and a slight melting due to the HHF test beyond the requirements. In addition to characterization of tungsten parts, the focus was set on the observation and properties of the cooling tube of both tungsten and CFC mock-ups. Interestingly, hardness measurement showed slight differences for the CuCrZr tube depending on the suppliers (manufacturing routes). Furthermore, micro cracks with a depth of up to 50 µm were found at the inner surface of the cooling tube with no appreciable consequences on the integrity of the cooling tube. This paper will present the factual observations and tentative explanations.

Id 629

Abstract Final Nr. P2.089

Impact on Tungsten of Transient Heat Loads on top of steady state Plasma Exposure

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High power loads during intense transient events such as bursts of Edge Localised Modes (ELMs) could pose severe threat causing erosion, melting and surface modifications of plasma-facing components (PFC) in fusion plants like ITER and DEMO. In addition to thermal shock events with power densities around 1 GW/m2, PFCs are subjected to high fluxes of energetic neutrons, hydrogen and helium ions. Under combined loading conditions, synergistic effects might lead to increased material damage during transients. Cracking thresholds and crack patterns in tungsten targets after repetitive ITER-like ELM heat pulses in combination with plasma exposure in PSI-2 (ƒ¡target=2.5.4.0 • ~1021m-2s-1, ion energy onto the surface Eion=60eV, Te.10eV) have been studied in recent experiments. The heat pulses were simulated by laser irradiation. A Nd:YAG laser capable of delivering up to 32J of energy per pulse with a duration of 1ms at the fundamental wavelength ƒÉ=1064 nm has been used to irradiate ITER-grade tungsten samples in the frequency range 0.5Hz-10Hz at room temperature (RT) as well as for targets preheated to 400□‹C . The observed damage threshold under pure heat loads for ITER-grade W lies between 0.38 and 0.76GW/m2 after 100 laser pulses. Cycling up to 1000 pulses at RT results in enhanced erosion of crack edges and crack edge melting. At the base temperature of 400□‹C, the formation of cracks is suppressed. In contrast to pure thermal exposure with laser beam, the experiments with pre-loaded W-samples as well as under combined loading conditions show a lower damage threshold of 0.3GW/m2 most probably due to hydrogen embrittlement and/or a higher defect concentration in a surface near region due to supersaturation with hydrogen. The observed surface roughening due to plastic deformation is also more pronounced. The impact of the surface modification of bulk tungsten on the fuel retention will be presented.

Id 887

Abstract Final Nr. P2.091

High Heat Flux testing of Tungsten Monoblock Mock-ups for the ITER Divertor

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With the aim to assess the option to start the ITER operation with a full tungsten divertor, an R&D program was launched in order to assess the performances of W armoured Plasma Facing Components under the conditions expected in the divertor target strike point region. The F4E program consisted in the manufacturing and High Heat Flux (HHF) testing of W monoblock mock-ups and medium scale prototypes up to 20 MW/m2. During the test campaign, twenty six W mock-ups and 2 medium scale prototypes manufactured by Plansee SE (A) by Hot Isostatic Pressing and by Ansaldo Nucleare (I) by Hot Radial Pressing have been tested at the FE200 (AREVA Le Creusot, France) and IDTF (Efremov Institute Saint Petersburg, Russian Federation) electron beam test facilities. The HHF testing program foresaw the performance of 5000 cycles at 10 MW/m2 and 300+700 cycles at 20 MW/m2 with 10 s power on and 10 s dwell time. The coolant conditions were representative of the Inner Vertical Target ones and a swirl tape (twist ratio = 2) turbulence promoter was provided. The test results fulfilled the qualification requirements (5000 cycles at 10 MW/m2 and 300 cycles at 20 MW/m2). Although a few mock-ups did not sustain the extended test program (additional 700 cycles at 20 MW/m2), the analysis of the results gave indications on potential improvements, in particular concerning the W material itself with the objective to remove the self-castellation of the W monoblocks and concerning the thermo-mechanical fatigue performances of the heat sink. In addition, some critical heat flux experiments, whose results confirmed those previously obtained were also performed. The main results will be presented and discussed in the paper.

Id 211

Abstract Final Nr. P2.092

High heat flux testing of ITER Normal Heat Flux First Wall (NHF FW) Mock-ups with calibrated defects

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The ITER NHF FW panels, able to withstand heat flux up to 2 MW/m2, are currently on-going the qualification activities in order to release the manufacturing of the 218 panels to be delivered by F4E. Within the framework of the development of acceptance criteria for the manufacturing, a series of High Heat Flux (HHF) tests of 2 sets of 7 FW mock-ups with Beryllium armour has been performed. Each set of tested mock-ups is composed of 6 mock-ups holding different types of embedded defects and 1 mock-up without calibrated defect. The HHF tests are performed in the Electron-beam test facility in Efremov Institute, Saint Petersburg, Russia. The test protocol includes 1000 heat cycles at 1.5 MW/m2 followed by successive steps of 200 cycles at heat fluxes increasing from 2 to 3 MW/m2. The aim is to highlight whether some of the embedded defects would generate an accelerated degradation of the Be/Cu bond and a sub-sequent possible overheating compared with the tiles without defects. The results of each series of tests are presented in detail and the impact of each kind of defect is analysed.

Id 266

Abstract Final Nr. P2.093

Progress in the design of Normal Heat Flux First Wall panels for ITER

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A typical NHF FW panel consists of a series of fingers, which represent the elementary plasma facing units and are designed to withstand 15,000 cycles at 2 MW/m2. The fingers are mechanically joined and supported by a back structural element or “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 218 NHF FW panels are divided in different main and minor variants. At present, more than 30 variants (main and minor) are foreseen, resulting in a significant effort for design and standardization of the NHF FW panels. The aim of this paper is to present the design work performed towards the PA signature. CAD detailed models have been created in CATIA for main and minor variants. Examples of local design solutions, as well as design work to achieve the global configuration of specific modules are provided. Finite Element (FE) simulations have been carried out, in order to simulate the operational scenario of the ITER and assess the thermo-mechanical behaviour of the most important FW panels against the required design criteria. This design and analyses activity is required to progress towards the finalization of the detailed design of the NHF FW panels.

Id 238

Abstract Final Nr. P2.094

Manufacturing and testing of a ITER First Wall semiprototype for EUDA pre-qualification

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This paper describes the main activities carried out in the frame of EU-DA prequalification for the supply of Normal Heat Flux (NHF) First Wall (FW) panels to ITER. It consists firstly in the manufacturing development, the fabrication and the factory acceptance tests of a reduced scale FW prototype (Semi-Prototype (SP)) of the NHF design. The SP has a dimension of 221 x 665 mm2, corresponding to about 1/6 of a fullscale panel, with six full-scale "fingers" and bearing a total of 84 beryllium tiles. It has been manufactured by the AREVA Company in France. Then, this SP is tested under high heat flux (HHF) in the dedicated test facility of Judith-II in FzJ, Germany. The objective of the HHF testing is the demonstration of achieving the requested performance under thermal fatigue. The manufacturing process has made extensive use of Hot Isostatic Pressing, which was developed over more than a decade during ITER Engineering Design Activity phase. The main manufacturing steps for the semi-prototype are described, with special reference to the lessons learned and the implications impacting the future fabrication of the full-scale prototype and the series which consists of 218 panels plus spares. As far as testing is concerned, the test protocol is presented and the behavior of the fingers under the 7500 cycles at 2 MW/m2 plus 1500 cycles at 2,5 MW/m2 is described in detail.

Id 588

Abstract Final Nr. P2.095

Geometry Sensitivity of Magnetohydrodynamic Flow with Flow Channel Insert Studied by Numerical Simulation

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Flow channel insert (FCI) made of a silicon carbide composite is a key element in the application of liquid metal blanket. The magnetohydrodynamic (MHD) effect and velocity distribution in a laminar liquid metal rectangular duct flow with an insulating FCI and pressure equalization slot (PES) subject to a uniform magnetic field is studied numerically by MHD module of Fluent. The electric current is solved by induced magnetic field formula. A very fine grid system is used to get a mass and electric current conservation resolution. Two cases with different pressure equalization slot widths are studied under low Hartmann number. The results show that the electric current distribution in the slot is influenced by the slot geometry dimension significantly. We find a butterfly shape electric current distribution in the slot for smaller pressure equalization slot case. This can explain the low velocity in the slot region well. The pressure equalization slot takes effect locally. The pressure gradient along the duct at different fluid region is different. Therefore there is a three-dimensional pressure distribution along the duct.

Id 682

Abstract Final Nr. P2.096

Fabrication of a semi-prototype of a Normal Heat Flux First Wall Panel for ITER

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This paper describes the activities performed by a Consortium of the 3 following companies: Iberdrola Ingeniería y Construcción, AMEC and Mecánica Industrial Buelna, under a contract from F4E, to develop and manufacture two small-scale mock-ups and a semi-prototype, which represents 6 full-scale fingers of a Normal Heat Flux (NHF) First Wall (FW) panel for ITER. These components consist of a bi-metallic structure, made of copper alloy bonded to a stainless steel base, on which beryllium tiles are joined. The different parts of the panel sub-components are produced by computer numerical control machining. The machining and the handling of beryllium have to be performed paying special attention to the health and safety requirements generated by the use of this material. The two above-mentioned joints are performed by Hot-Isostatic Pressing (HIP). This process uses high temperature and pressure, generated by an inert gas, usually Argon, to bond the different parts of the component by diffusion, according to the two following HIP cycles: - First HIP cycle: in which SS and CuCrZr parts are joined together to form the heat sink, at 1040ºC and 140MPa - Second HIP cycle: in which the beryllium tiles are bonded onto the copper alloy surface of the heat sink, at 580ºC and 140MPa The HIP process requires previous operations to facilitate the bonding (chemical cleaning, ion coating, electroplating, etc.) and heat treatments in order to obtain good final mechanical properties of the HIP bonded structures. The FW panel complex geometry, together with the advanced manufacturing processes needed for its fabrication, presents several technological challenges that are explained in detail in this paper.

Id 504

Abstract Final Nr. P2.097

Preliminary numerical simulation of buoyancy effects of mhd flow for SLL-TBM

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The liquid lithium-lead(PbLi) breeder blanket concept is an attractive candidate for the test in ITER. To check and validate the feasibility, the China dual-functional lithium lead test blanket module(DFLL-TBM) system is designed to demonstrate the integrated technologies of both He single coolant(SLL) and He-PbLi dual-coolant(DLL) blankets. In SLL blanket, the lithium lead is quasi-static and just considered as tritium breeder, the heat in the breeding region is cooled by the helium. An important aspect of MHD flows in SLL-TBM is related to non-uniform volumetric heating by fusion neutrons causing buoyant flows, which may influence or even dominate the performance in SLL-TBM where the breeder/coolant forced flow is quasi-static. In these conditions, the buoyancy forces affect the velocity profiles significantly and create recirculating flows, whose magnitude of velocity can even overcome that of the forced flow. In this paper the main characteristics of buoyancy effects of MHD flows in poloidal duct in SLL-TBM are described numerically, and the mechanism of buoyant convection in the presence of an imposed magnetic field is studied to understand the liquid metal circulating in the blanket. Effects of the direction of the heat flux with respect to the orientation of the magnetic field and the influence of electric conductivity of walls on the flow structure are investigated. Of particular interest is the formation in boundary layers of velocity jets driven by temperature gradients.

Id 785

Abstract Final Nr. P2.098

New design aspects of cooling scheme for SST-1 plasma facing components

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Plasma facing components (PFC) of SST-1 comprising of baffles, divertors and passive stabilizers have been designed and fabricated for a maximum heat load up to 1.0 MW/m2. In operational condition, SST-1 divertors and passive stabilizers are expected to operate with a heat load of 0.6 and 0.25 MW/m2. Each PFC is made up of copper alloy (i.e. CuCrZr and CuZr alloy) back plate on which graphite tiles have been attached mechanically. Additionally soft graphoil sheets of 0.7 mm have been sandwiched between copper back plate and graphite tiles in order to increase the area of contact. During plasma operation, the heat loads on PFC are required to be removed promptly and efficiently. Thereby the design of an efficient cooling scheme becomes extremely important for an efficient operation of PFC. PFCs are also baked up to 350ºC in order to remove absorbed moistures and other gases. 3D thermal analyses of PFC using ANSYS have been carried out to ensure its thermal stability. The cooling parameters have been calculated according to average incident flux on divertors and passive stabilizers. The flow behavior has been estimated using incompressible pipe flow analysis and system modeling software AFT Fathom. This software ensures the coolant flow parameters and behavior in each module of PFC in assembled state. Based on AFT Fathom results final cooling parameter and structuring of cooling line assembly for PFCs are planned. Engineering design demonstrated the required mass flow rate and velocity for cooling water in each sub-connection are optimized to be 0.43 kg/s and 5.5 m/s for efficient heat extraction under steady state heat load. Maximum temperature which PFC could be maintained is 355ºC and is well within threshold limits of material property degradation. The header distribution, modeled using AFT Fathom, resulted for required parameters within maximum 5% of deviation.

Id 57

Abstract Final Nr. P2.099

Development of laser induced breakdown spectroscopy for studying erosion, deposition, and fuel retention in ASDEX Upgrade

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Laser induced breakdown spectroscopy (LIBS) is a promising method for the remote in situ monitoring erosion/deposition processes and retention of plasma fuel on the first walls of fusion reactors. Post-mortem LIBS analysis of ASDEX Upgrade samples exposed to plasma during 2009 experimental campaign was carried out. The analysed samples were taken from tiles belonging to different characteristic regions (erosion dominated, shadowed etc.) of the device. Originally, before exposure to plasma the samples had 1-10 m W coating on graphite. Nd:YAG laser operating at 1064 nm produced the plasma plume of the sample material. Argon as a background gas was used at pressures of 10-4 and 100 Pa. Using optimized recording parametres, time resolved single shot LIBS spectra were recorded simultaneously in two different directions. In the direction collinear with the laser beam the spectrum was recorded in 250-850 nm wavelength range by Mechelle 5000 spectrometer coupled with Andor ICCD camera. Another spectrometer arrangement (MDR-23 spectrometer with Andor iStar camera) with higher spectral resolution and sensitivity recording spectra in the side direction was used for deuterium detection at the spectral range around the D line at 656.1 nm. The LIBS results were compared with data obtained by three different ion-beam techniques: Rutherford backscattering, nuclear reaction analysis, and secondary ion mass spectrometry . Using LIBS spectra as a function of laser shot number, elemental depth profiles of main components (W, C, B,) of the layers were deduced. The LIBS profiles matched qualitatively with profiles obtained by SIMS. Because of Stark broadening and the comparatively intensive Hline (likely caused by surface water), the reliable detection of deuterium and therefore building its depth profile was possible only in the case of few samples. Comparison with NRA measurements gave the lower limit of the surface density of deuterium which could be detected by LIBS.

Id 714

Abstract Final Nr. P2.100

Retention behaviour of deuterium in beryllium under single D+ and dual He+/D+ exposure

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We have studied the effects of helium irradiation in the retention mechanism of deuterium implanted in beryllium. For this purpose, four batches of beryllium plates were bombarded with single 15 keV D+ and dual 30/15 keV He+/D+ ion beams at two different fluences (1 and 5e17 at/cm2, under and above the saturation limit for deuterium). The samples were annealed afterwards at 523, 723 and 923 K for 10 min. The retained amounts of deuterium and helium, the formation of beryllium hydride (BeD2) and the morphological changes caused by ion implantation and annealing were evaluated by means of nuclear reaction analysis (NRA), X-ray diffraction (XRD) and scanning electron microscopy (SEM), respectively. Almost all the impinging He ions became retained in the as-deposited surfaces, while the corresponding retention for deuterium were significantly lower and decreased under helium exposure. After heating, both helium and deuterium contents decreased, and at 923 K they were no longer observed in the NRA spectra. A BeD2 pattern was never identified by XRD. The morphology of the irradiated surfaces remained unchanged after single D+ implantation and annealing up to 723 K. Nevertheless, a smooth blistering effect was already visible at 923 K. In opposition to this behaviour, the dual D+/He+ implantations induced a strong blistering mechanism by annealing at 723 and 923 K. Additional thermal desorption spectroscopy (TDS) measurements were performed with the purpose to investigate the binding states of the retained species. After single D+ exposure with a fluence of 5e17at/cm2, the release spectrum revealed a retention behaviour in ion induced trap sites and a supersaturation of the beryllium bulk. For dual He+/D+ implantations the results suggest the mitigation of the saturation mechanism and the rise of the binding energy states of deuterium.

Id 638

Abstract Final Nr. P2.101

Status of the beryllium tile bonding qualification activities for the manufacturing of the ITER first wall

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The first wall (FW) is the main protection of the ITER vessel against the plasma, and must accommodate heat fluxes from 2 to 4.7 MW/m². It is made of 440 panels with typical dimensions of 1.5 m toroidally and 1 m poloidally, cladded with 8-10 mm thick beryllium (Be) armour tiles of 16 to 42 mm in size. The FW has passed successfully its final design review in 2013. It is now in a pre-manufacturing phase with the preparation and signature of procurement arrangements. The FW is procured by EU (50%), RF (40%) and CN (10%) Domestic Agencies. The FW manufacturing involves critical manufacturing steps which are monitored through the qualification stages. One is the bonding of the beryllium tiles to the copper (Cu) alloy cooled structures, since this material pair has dissimilar thermo-physical properties and the phase diagram contains brittle intermetallics. The tile bonding joint must withstand stresses close to or beyond the yield strength during 15,000 thermal cycles. The qualification program involves heat flux testing of semi-prototypes, with the aim of demonstrating successful bonding for a large number of tiles (90% of the surface) at up to 125% of the design heat load. In parallel to the qualification, additional R&D activities and investigations are undertaken, aiming at a refined characterisation of the tile bonding behaviour and definition of acceptable bond defects for the manufacturing stage. Bonding defects are observed to be more resilient to propagation under heat flux cycling, compared to the previous large manufacturing experience obtained with carbon composite (CC) tiles. Thermo-mechanical analyses shows that better matched material properties and lower heat load cause lower stress intensity factor for the Be/Cu pair compared to the CC/Cu one. Similarly to the CC/Cu pair, a key factor governing bond damage appears to be the operation temperature of the bond.

Id 332

Abstract Final Nr. P2.102

Development of Residual Thermal Stress-Relieving Structure of CFC Monoblock Target for JT-60SA Divertor

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Carbon Fibre-reinforced Carbon (CFC) monoblock target for JT-60SA divertor is under development toward mass-production of 1,000 targets. A CFC monoblock, a CuCrZr cooling tube at the centre of the monoblock and a WCu interlayer were bonded by vacuum brazing in a high temperature, into a target. After the bonding, strong tensile stress was generated in the CFC monoblock around the CuCrZr cooling tube in a room temperature condition due to difference of thermal expansions between CFC and CuCrZr. If the stress exceeds the maximum allowable stress of the CFC, cracks are generated in the CFC monoblock and heat removal capacity of the target degrades. In the previous trial productions, nearly 200 targets were manufactured. Only 100 targets passed the acceptance test, and the rest of them showed significant degradation of heat removal capacity. In this research, a new structure of the targets was proposed, to reduce residual thermal stress and to depress the degradation of heat removal capacity of the targets, toward the mass-production. The following measures were implemented on the proposed structure; - Replacement of the interlayer material (WCu -> OFCu) - Slitting in the CFC monoblock aside of the cooling tube - Shifting of the cooling tube (at the centre of the monoblock -> near plasma facing surface) The effectiveness of the measures were evaluated by numerical simulations. The averaged residual thermal stress in the CFC monoblock at the front side of the bonding face, Ef, was employed as an evaluation scale. The proposed measures could drastically reduce the averaged stress Ef from 106 MPa with the conventional structure to 30 MPa. Target mock-ups with the proposed structure were manufactured. Infrared thermography inspection and high heat flux test were carried out on the mock-ups in order to evaluate thermal performance of the mock-ups.

Id 320

Abstract Final Nr. P2.103

Progress of ITER Full Tungsten diveror technology qualification in JAEA

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Japan Atomic Energy Agency (JAEA) is in progress for technology qualification toward Full-tungsten (W) ITER divertor outer vertical target (OVT), especially, tungsten monoblock technology that needs to withstand the repetitive heat load as high as 20 MW/m2. To demonstrate the armor heat sink bonding technology and heat removal capability, 6 small-scale W monoblock mock-ups manufactured by different bonding technologies using different W materials in addition to 4 full- scale prototype plasma-facing Units. After non-destructive test, the W components were tested under high heat flux (HHF) in ITER Divertor Test Facility (IDTF) at Efremov Institute, St Petersburg Russia. Consequently, all of the W monoblocks endured the repetitive heat load at 20 MW/m2 for 1,000 cycles (requirements 20 MW/m2 for 300 cycles) without any failure such as detachment of the W monoblocks, degradation of the heat removal capability and water-leak from the cooling tube. This demonstrated the availability of W monoblock technology in Japan. In addition to the armor to heat sink joints, the load carrying capability test on the W monoblock with a leg attachment was carried out. In uniaxail tensile test, all of the W monoblock attachments with different bonding technologies such as brazing and HIPping withstand the tensile load exceeding 20kN that is the value more than twice the design value. The failures occurred at the leg attachments or the W monoblocks, rather than the bonding interface of the W monoblocks to the leg attachment. In this paper, the details of test results, and the correlation of HHF test results and non-destructive test will be reported.

Id 460

Abstract Final Nr. P2.104

Measuring robustness of maintenance schedules in Fusion Remote Handling

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Remote handling operations during a planned maintenance shutdown of a fusion plant like JET or ITER are executed according to formally described and validated procedures. Unexpected events during the execution of the procedure may incur delays or require rescheduling, suspending the execution of the current procedure to pick it up later. The delayed procedure can continue to block access to reserved tooling, or temporarily release them for use in another task. Depending on the way it is done, this may in turn influence other procedures, and hence affect the effectiveness of the maintenance schedule for the entire shutdown period. Taking proactive countermeasures against these unexpected events in the planning aids the maintenance schedule accuracy. In this paper, we investigate whether adding certain amounts of slack influences the robustness of the overall shutdown schedule. Slack is defined as additional time allocated to the execution of tasks. It serves to absorb delays and minimizes disruptions. For the analysis of typical anomalies that occur in fusion plant maintenance, we analyse the execution of a complex maintenance procedure in JET: The installation of the ITER-Like Wall (ILW), where several unforeseen challenges were encountered. Factors affecting the ratio between estimated versus actual durations for operations have been described. We introduce a classification for different types of anomaly that may occur during execution of operations and propose metrics to measure the robustness of a remote maintenance schedule against various types of disruption. Using simulation, we inspect the occurrence of disruptions in a set of procedures, and evaluate the effect on the overall schedule. Finally, recommendations are made as for how much slack is appropriate for robustness required for the maintenance schedules.

Id 617

Abstract Final Nr. P2.105

Mapping ITER Port Plug maintenance workflow

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This paper presents the results of an effort to map all maintenance activities related to the ITER EC Upper Launcher (UL) port plug in a generic workflow. The maintenance activities include the assembly and disassembly of interfaces and auxiliary equipment; hands-on maintenance inside the Upper Port Cell; logistics between the Tokamak and the Hot Cell Facility (HCF); Remote Handling (RH) Operations inside the HCF; cleaning and disposal of waste and; testing of the UL in the Port Plug Test Facility (PPTF). Maintenance activities on any ITER in-vessel component can have far reaching consequences. Besides the obvious influence on machine availability, there are aspects like tool certification, procedure validation, operator training, spare part inventory, the amount of waste, and post maintenance system performance. These aspects are interdependent in an environment where the optimal maintenance strategy is difficult to determine on beforehand. The paper shows that relatively simple administrative methods in combination with interactive analysis based on Virtual Reality (VR) can provide a grip on this complex matter. With a small set of RH Plant and Task Definition Forms we describe a key subset of the activities surrounding plug maintenance. The VR analysis adds in-depth insights, which dramatically improve the effectiveness and quality of the documentation, including the generation of tooling and procedure overviews. We also show how the introduction of RH specific design features and standardization of interfaces can contribute to a significant reduction of maintenance time, reduce waste production and put a limit on tool proliferation, without negatively affecting the system performance.

Id 701

Abstract Final Nr. P2.106

Design assessment of triangular support bracket for manufacturability

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The triangular support is connected structurally and hydraulically with the inner shell of the vacuum vessel and its main role is to keep plasma vertical stability during operational disruptions. Korea is responsible for the procurement of sectors 1 and 6 of the main vessel including triangular support. At present, design review for its fabrication by ITER Korea and Hyundai Heavy Industries, Co., Ltd. is in progress. This paper presents the results on various designs for triangular support bracket in terms of manufacturability considering both easiness of non-destructive evaluation and fabrication efficiency. The several designs are proposed and evaluated under the most critical loading condition using elastic and limit analysis with fatigue evaluation. Consequently, an optimized design in structural safety is determined in accordance with RCC-MR. This result is reconfirmed by cross-checking in comparison with the baseline of integrity that already had been determined by IO. The design deviation requests of triangular support bracket have been submitted to ITER Organization and ANB (Agreed Notified Body) for approval, and their verification is currently under discussion.

Id 326

Abstract Final Nr. P2.107

CFD meshing methodology for large computational domains applied to an irregular sector of the ITER vaccum vessel.

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A 3D CFD thermal hydraulic analysis of the irregular sector #2 (IrS#2) of the ITER vacuum vessel was carried out with the commercial software ANSYS FLUENT®. In the present work, the meshing methodology, which is critical for the reliability and quality of the results, is exposed. Large computational domains like that of the IrS#2, with 12m high to 7m width, pose a problem not only from the computational resources standpoint, but also from the procedure to build the mesh. The absence of symmetries, the complexity of the geometry, the fact that solid and fluid domains must be coupled to account for thermal behavior and the resolution needed to generate high quality results, makes impossible the implementation of common meshing techniques. In addition, neither hexahedral nor tetrahedral meshes may be applied due to either the type of geometry or convergence issues in the CFD calculations. In this work a parallel working procedure together with a polyhedral based meshing procedure is exposed so as to show how large computational domains can be meshed without neglecting the physical phenomena that takes place within complex fusion technology related components. The procedure shows the insight acquired, from the CATIA geometry healing and simplification to the polyhedral mesh conversion and parallelization, during the CFD analysis of the IrS#2.

Id 132

Abstract Final Nr. P2.108

New portable machine for the in-situ inspection, repair and manufacturing of complex features in remote locations within the vacuum vessel of ITER.

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Last years have seen an increasing demand for large and precise parts following the growth of several sectors, such as conventional and nuclear power plants, shipbuilding industry, railroad, offshore platforms, large scientific facilities like ITER and others. These sectors not only require high precision manufacturing of large parts, but also maintenance and repairing operations of complex features during the assembly operations, in configurations and situations of difficult access and challenging set-up. This paper proposes a change in the paradigm to tackle the manufacturing and maintenance of these large parts: to dispose the dogma “work piece inside machine” in the given case and replace it by the principle “small machines on large work pieces”. That change leads to the study and development of small mobile machines that can move along large parts to perform the required operations. These machines are conventionally called portable machines. This paradigm makes the machine closer to a free robot than a conventional machine and thus new needs appear for external referencing (navigation) and internal referencing (feature recognition within the workpiece). Following this philosophy and aiming at the preparation, inspection and recovery of welding seams within the ITER vacuum vessel, this paper presents a new five-axis, miniature milling machine based on a serial kinematic architecture, with remarkably homogeneous stiffness behavior for every possible machine orientation and process combination. This portable machine can perform both mid-duty milling and drilling operations in a five axis configuration and is equipped with a laser scanner with a typical precision of 50 µm, combined with a touch probe for redundancy and finer data acquisition.

Id 956

Abstract Final Nr. P2.109

Human-Robot Interface architecture for a Multi Purpose Rescue Vehicle for remote assistance in ITER

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The Remote Handling (RH) is a requirement in nuclear test facilities, namely for vessel transportation operations. In ITER, the Cask and Plug Remote Handling System (CPRHS) is an essential large vehicle for the remote transfer of clean and activated components between the Hot Cell and Tokamak buildings. The CPRHS is composed by the cask envelope, the Cask Transfer System (CTS), which acts as a mobile robot and the pallet. In case of CPRHS or CTS failure during the maintenance activities, rescue and recovery operations must take place. Since no human being is allowed during the RH activities, a Multi-proposed Rescue Vehicle (MPRV) must be available for providing support in site. A proposed MPRV is equipped with different sensors, such as video cameras, laser range finders, thermometers, probes, dosimeters, etc. In addition, it is also equipped with actuators, namely robotic manipulators with different end effectors and a wheeled system to move the entire body. The sensors can be articulated with the manipulators to maximize the coverage area. The MPRV must be remotely operated, which becomes a challenge for the human operator, given the large amount of information and the high number of degrees of freedom. This paper proposes a human-robot interface to operate a MPRV, aiming at an immersive and tele-operated environment to enhance the situation awareness of the operator and to maximize all the MPRV capabilities for unexpected situations. The interface is designed with a combination of augmented reality together with haptic technologies, following a user-centered design principle. In addition, the system includes a surveillance system to help the human decision-making, to mitigate the human error and to monitoring the MPRV performance within the scenario.

Id 789

Abstract Final Nr. P2.110

Assessment of ex-vessel movers in Remote Maintenance Systems of DEMO

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The Remote Maintenance System (RMS) is a key issue in any nuclear fusion power plant during various planned and unplanned maintenance operations. The plant architecture for RMS must allow a trade-off between the optimization of plant maintainability and other tradable factors, namely system performance, cost, structural efficiency, performance and configuration of other systems. In particular it is applicable for DEMO, which is intended to be a demonstration of a power plant that lies between the ITER and a 'first of a kind' commercial station. The main RMS activities in the DEMO Active Maintenance Facilities (AMF) involve cranes and transfer casks. To proceed with the AMF design, it is crucial to address questions related with these ex-vessel movers. This work reports the first studies of some of these issues, such as the processes of transportation, timing and logistics; number of cranes and casks and their typologies; feasibility analysis given the cranes and casks envelopes and the cask kinematics along optimized trajectories based on the AMF CAD models; risk analysis of clashes and failure; rescue and recovery operations and conflicts with doors profiles. For the RMS activities, an overhead crane is considered, as a girder underslung crane type equipped with a lift system. For the transfer cask a solution is proposed based on the Cask and Plug Remote Handling System (CPRHS) of ITER: a transport system and the cargo over a pallet. The kinematics of the transport system and the envelope are proposed, which are crucial for the feasibility analysis of the AMF design. A software tool, specifically developed during previous projects related with RMS in ITER, is used to carry out the feasibility. This paper presents a summary of the so obtained results, draws conclusions on the AMF design and points out considerations in and beyond the RMS activities.

Id 563

Abstract Final Nr. P2.111

Assessment and performance optimization of the ITER plasma position reflectometry in-vessel oversized waveguide bends

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The ITER plasma position reflectometry (PPR) diagnostics system comprises five reflectometers distributed both toroidally and poloidally at four locations, commonly known as gaps 3 to 6. The system will operate in O-mode in the frequency range 15 to 75 GHz. The primary function of the PPR system is to provide a stand-by measurement of the distance (gap) between the first-wall and a given density layer, provided by the plasma control system. Here we'll consider only the reflectometers located at gaps 4 and 6, commonly known as in-vessel systems. These systems access the plasma through small apertures between vertically adjacent blanket modules: gap 4 is located on the low-field side entering the vessel through Upper Port 1, while gap 6 probes the plasma from the high-field side and accesses the vessel through Upper Port 14. To minimize losses, the in-vessel transmission lines of gaps 4 and 6 use slightly oversized rectangular waveguides whose outer dimensions (22x14 mm) are imposed by space availability. Moreover, as the waveguides are welded to the vessel wall its routing has to adjust to the vessel geometry. To cope with these constraints, the waveguide layout features two 90º bends, behind the blanket modules, just before/after the launching/receiving antenna, both on gaps 4 and 6. In addition, gap 4 features two 120º bends located at the vessel-to-port transition. Oversized waveguide bends are critical components known to excite higher frequency modes, which can lead to significantly higher losses making it important to assess and optimize its performance. Here we'll present the results obtained from the simulations performed using HFSS to assess the electromagnetic performance of both bends. The performance optimization achieved by changing the geometry of the bends as allowed by space and geometry restrictions is also presented.

Id 633

Abstract Final Nr. P2.112

Performance assessment of the iter plasma position reflectometry in-vessel antenna setup

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The ITER plasma position reflectometry (PPR) diagnostics system comprises five reflectometers distributed both toroidally and poloidally at four locations, commonly known as gaps 3 to 6. The system will operate in O-mode in the frequency range 15 to 75 GHz. The primary function of the PPR system is to provide a stand-by measurement of the distance (gap) between the first-wall and a given density layer, provided by the plasma control system. Here we consider only the reflectometers located at gaps 4 and 6, commonly known as in-vessel systems. These systems access the plasma through small apertures between vertically adjacent blanket modules: gap 4 is located on the low-field side entering the vessel through Upper Port 1, while gap 6 probes the plasma from the high-field side and accesses the vessel through Upper Port 14. Inside the vessel, due to space and geometry restrictions and to the need to protect the front-end antennas from the nuclear heating both systems view the plasma through small apertures between vertically adjacent blanket modules. The design of the antennas is strongly influenced by the need to maximize the coupling to the plasma and by the need to cope with the high nuclear heating and with the high mechanical stress due to the strong electromagnetic forces arising during fast transient events, like plasma disruptions. The antenna configuration proposed in the original design featured two parallel asymmetric aperture horns. Currently, a design identical to the one proposed to the high-field side main profile reflectometry system is being considered, which uses a higher gain launching horn together with a two-mirror receiving arrangement. In this paper we present the results obtained from simulations performed using HFSS to assess the performance of both the original and current antenna setups.

Id 987

Abstract Final Nr. P2.113

A large scale divertor manipulator for ASDEX Upgrade

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ASDEX Upgrade (AUG) is a mid-size tokamak fusion experiment. In 2013 a new divertor, Div-III, was installed replacing the tungsten coated target tiles in the outer divertor by solid tungsten plates. During the concept and design phase of Div-III the option of flexible divertor instrumentation and divertor modification as contribution for divertor investigations in preparation of ITER was given a high priority. In order to keep the operational space and to gain flexibility for divertor modifications a large scale divertor manipulator, DIM-II, was designed and installed. DIM-II allows to retract 2 out of 128 outer divertor target tiles including the water cooled support structure into a target exchange box and to replace targets without breaking the vessel vacuum. DIM-II is based on a carriage-rail system with a driving rod pushing a front-end into the divertor position for plasma operation. Three groups of front-ends are foreseen for physics investigations: (i) modified standard targets clamped to the standard cooling structure, (ii) dedicated front ends making use from the whole available volume of about 200x160x80 mm3 and (iii) actively cooled targets. For all three groups front-ends are in preparation, e.g. castellated targets, marker probes with gas puffing and a rotatable deposition probe allowing time/shot resolved measurements. Electrical probes such as ExB or a retarding field analyzer are under consideration. The thermal properties of actively cooled targets can be tested in high heat flux test beds, such as GLADIS. Effects of magnetized plasmas with a shallow angle of incidence can be investigated in plasma devices only. DIM-II allows exposing actively cooled targets with up to 230°C temperature of the cooling water. This paper will present the DIM-II design including the FEM based modifications of the divertor support structure and the front end options and will outline the operational range of the different front-end groups.

Id 880

Abstract Final Nr. P2.114

Fabrication Feasibility Study on Copper Cladding in Tokamak System

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To improve the plasma vertical stabilization during Tokamak operation, partial pure copper layer was considered on the inner shell of vacuum vessel which is made of stainless steel. However, the copper cladding on formed plates has been a great challenge due to the welding joints of large complicate component and the difficulty of dissimilar metal joining. From an engineering point of view, thin copper layer has to be coated after welding assembly of several parts with tight tolerance requirement. There are several candidates for the copper cladding such as cold spray coating, overlay welding and explosive welding. In this paper, fabrication feasibility study of copper cladding was carried out by using overlay welding and cold spray method. Microstructural characteristics and mechanical properties have been investigated. In case of the overlay welding, the copper infiltration into stainless steel and many cracks on the welded joint were observed. Whereas, the cold sprayed joints had no delamination and no cracks. Therefore, the copper cladding on a full scale mock-up using cold spray method had performed to verify the fabrication feasibility. The thickness and the average surface roughness (Ra) of copper layer on the mock-up are about 3.3 mm and 3.84 μm, respectively. These are well satisfied with the technical requirements of copper cladding. The results of liquid dye penetration test shows that no defects were observed on the surface of cold sprayed copper layer. At present, the copper cladding on vacuum vessel has not been adopted due to the interface issues. Nevertheless, the fabrication feasibility results of the copper cladding using cold spray method could be contributed to the design development of fusion devices because of their excellent achievement. This paper will present the preliminary and mock-up test results of the copper cladding.

Id 495

Abstract Final Nr. P2.115

Comparative evaluation of Remote Maintenance Schemes for Fusion DEMO Reactor

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Maintenance schemes are one of the critical issues in DEMO design, significantly affecting the configuration of in-vessel components, the size of toroidal field (TF) coil, the arrangement of poloidal field (PF) coils, reactor building, hot cell and so forth. Therefore, the maintenance schemes should satisfy many design requirements and criteria to assure reliable and safe plant operation and to attain reasonable plant availability. Previously, compact DEMO, SlimCS designed by JAEA adopted a horizontal sector transport maintenance scheme taking account of high availability. In the sector maintenance scheme, the number of piping cutting/re-welding points is minimized. On the other hands, the sector transport maintenance scheme has disadvantages of larger TF coil and larger maintenance port, which has big impact on medium size DEMO (plasma major radius Rp ~8m). In order to decide a most probable remote maintenance scheme for DEMO, comparative study of maintenance schemes taking account of not only plant availability but also engineering reliability is important. In this study, we categorize various schemes in term of 1) the maintenance port position for transporting blanket segments, 2) blanket segmentation, and 3) divertor segmentation. The design study clarifies some assessment factors on DEMO remote maintenance scheme, for example, (1) minimization of the size and magnetic stored energy of TF coil and PF coils, (2) divertor maintenance, and (3) define the transferring mechanism in the vacuum vessel. In reviewing these assessment factors, the separated sector transport using the vertical maintenance ports with small divertor cassette maintenance scheme was found to be a more probable maintenance approach. This presentation describes engineering design of each maintenance schemes and evaluation results of comparison among maintenance schemes.

Id 693

Abstract Final Nr. P2.116

Development of remote pipe welding tool for divertor cassettes in JT-60SA

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Remote pipe welding tool accessing from inside pipe has been newly developed for JT-60SA. Remote handling (RH) system is necessary for the maintenance and repair of in-vessel components such as lower divertor cassettes in JT-60SA. Cooling pipes, which connects between the divertor cassette and the vacuum vessel with bellows are required to be cut and welded in the vacuum vessel by RH system. The available space for RH system is very limited inside the vacuum vessel, especially around the divertor cassettes. Thus, the cooling pipes are required to be cut and weld from the inside in the vacuum vessel. The inner diameter, thickness and material of the cooling pipe are 54.2 mm, 2.8 mm and SUS316L, respectively. The welding tool is vertically inserted into the pipe from the divertor cassette side. The outer diameter and length of the welding tool head are 36 mm and about 1m, respectively. The tool is able to weld a joint which locates 480 mm in depth from the mounting surface on the divertor cassette. The tool performs circumferential laser welding by rotating the tool inside the pipe. The tool was slightly tilted during welding, and the opposite side of laser emission hole of the tool was kept in contact with the inner surface of the pipe. An accuracy of laser spot during rotating the tool is less than ±0.1mm. An upper pipe connected to the divertor cassette has a jut on the edge to fill the gap between pipes. In addition to the jut, a wide welding bead is necessary to extend an allowable gap. Second circumferential welding was carried out after changing the positions of laser spot in order to obtain wide welding bead. Owing to the jut and two-times welding, the welding tool achieved the maximum allowable gap is 0.7mm.

Id 868

Abstract Final Nr. P2.117

Laser welding development to expand allowable gap in bore welding for ITER Blanket hydraulic connection

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Shielding blanket in ITER has an active cooling structure and needs hydraulic connections to the cooling water manifold. To replace the blanket, cutting and welding the hydraulic connection are necessary. Access for these operations is limited in a small hole in the first wall because of spatial constraint related to neutron and heat fluxes. After D-T experiments, these operations have to be remotely done, therefore, bore cutting and welding tools compatible with remote handling technique are inevitably required. JAEA had developed a bore tool of the laser welding, and pipes 42.2 mm in inner diameter and 2.77 mm-thick were successfully welded. In the test, the allowable initial gap between pipes was 0.2 mm. Relaxation of the tight requirement in the allowable gap is preferable in comparison with the assembly tolerance. This paper presents recent results of the laser welding tests where groove geometry was changed to expand the allowable initial gap. Additional material is required to avoid excess concavity caused by the initial gap between pipes. Generally wire fill is applied. However, this leads to additional complication in the remote handling tool. In the present study, the groove was machined to be partially thick instead of the filler wire. Firstly, plates with the partially thick grooves were welded to study preferable groove geometry and welding conditions. For the groove with additional metal 0.7 mm-thick and 4.0 mm-wide, the plates with the initial gap of 1.0 mm can be welded. Then the groove geometry and welding conditions were modified through the pipe welding tests. Thicker groove requires higher heat input, and increase of the heat input leads to burn through especially for the pipe welding. By the application of the additional metal 0.5 mm-thick and 2.5 mm-wide in the groove, allowable initial gap increased from 0.2 mm to 0.7 mm.

Id 866

Abstract Final Nr. P2.118

Welding technology on sector assembly of the jt-60sa vacuum vessel

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The JT-60SA project is conducted under the BA Satellite Tokamak Programme by EU and Japan, and the Japanese National Programme. The project mission is to contribute to early realization of fusion energy by supporting ITER and by complementing ITER with resolving key physics and engineering issues for DEMO reactors. In January 2013, the assembly of the JT-60SA started with cryostat base manufactured by Spain. Assembly of the vacuum vessel (VV), which is a main component manufactured by Japan, will start in May 2014. The vacuum vessel (150 tons) is a double wall torus structure and the maximum major radius of 5.0 m and height of 6.6 m. The manufacturing design concept is that the vessel is split in the 10 toroidal sectors manufactured at factory, and assembled on-site; seven of the 40-degree sectors, two of the 30-degree beside final one, and the final of the 20-degree. The final sector is assembled with the VV thermal shield and toroidal field magnets into the 340-degree as prepared in one sector. Sectors are temporally fitted on-site and adjusted one over the other before the assembly. After measurement of the dimensions and the reference, these sectors are transferred onto the cryostat base. First, three 80 degree sectors are manufactured with mating each 40 degree sector by direct joint welding. The rest sectors including the final sector are jointed with splice plates. Welding manipulator and its guide rails are used for these welding. In this paper, the detail of the VV sectors assembly including the final sector is explained. Welding technologies to joint the two of 40-degree sectors are reported with the present manufacturing status and the welding trial on the vertical stub with the partial mock-up of the final sector are discussed with the assembly process.

Id 943

Abstract Final Nr. P2.119

Safety Analysis of LOCA Accidental for the DEMO Blanket

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Loss of Coolant Accident (LOCA) caused by a pipe break of Helium Coolant System (HCS) is considered as one of the most critical accident for the Helium Cooled Pebble Beds Blanket system (HCPB). The consequent rapid helium blow-down requires a prompt plasma shutdown to prevent in-vessel components over-heating and failure. Even after the plasma shutdown the temperature can increase over the design limit due to decay heat. The temperature increase of a blanket vertical segment due to decay heat is an important parameter required in the design of the blanket attachment with the Vacuum Vessel (VV). Six cases have been studied taking into account both ex-vessel and in-vessel LOCA. The DEMO HCPB Outboard Blanket (OB) temperature behaviour postulating full blanket loss of coolant event has been assessed with ANSYS code. The decay heat has been assumed as function of heat power in normal operation. The most challenging case has been selected to perform a thermal stress calculation to evaluate the maximum loads and displacement within the module and its connection to the Back Supporting Structure (BSS). The paper shows the results of the performed thermal analyses and consequent stress assessment in the blanket strcutures.

Id 234

Abstract Final Nr. P2.120

Design validation of the ITER EC Upper Launcher according to codes and standards

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The ITER EC Upper launcher has passed the CDR (Conceptual Design Review) in 2005 and the PDR (Preliminary Design Review) in 2009 and is in its final design phase now. The final design will be elaborated by the European consortium ECHUL-CA with contributions from several research institutes in Germany, Italy, the Netherlands and Switzerland. Within this consortium KIT is responsible for the design of the structural components (the Upper Port Plug) and also the design integration of the launcher. As the selection of applicable Codes and Standards was under discussion for the past decade, the conceptual and the preliminary design of the launcher structure were not elaborated in straight accordance with a particular code but with a variety of well-acknowledged engineering practices. For the final design it is compulsory to validate the design with respect to a typical engineering code in order to be compliant with the ITER quality and nuclear requirements and to get acceptance from the French regulator. This paper presents design verification of the closure plate, which is the vacuum and Tritium barrier and thus a safety relevant component of the Upper Port Plug, performed with the ASME Boiler & Pressure Vessel Code. Rationales for choosing this code are given as well as a comparison between different design methods, like the “design by rule” and the “design by analysis” approach. The selections of proper load specifications are covered as well as the identification of potential failure modes. In addition to that stress categorizations, analyses results, manufacturing aspects, required inspection, examination and testing procedures, and prove of the soundness of the design are presented.
Acknowledgement: This work was supported by the European Joint Undertaking for ITER and the Development of Fusion Energy (Fusion for Energy) under contract No. F4E-2010-GRT-161.

Id 513

Abstract Final Nr. P2.121

Advancement in HCPB DEMO Blanket design

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A HCPB blanket design development study has been performed at the Karlsruhe Institute of Technology within the frame of the EFDA PPPT 2013 activities for the DEMO vessel/in-vessel components. The main purposes were to integrate a dual cooling helium system within the module and to implement two attachment systems: a fixed supporting system for the blanket modules connection with the supporting structure and the attachment between the blanket segment and the vacuum vessel. A parametric CATIA model of the blanket box, the breeder units and the back supporting structure (BSS) has been realized to enhance the flexibility and, hence, more possibility for optimization of the design through compliance to different requirements, such as mechanical loads, Tritium Breeding Ratio (TBR), and safety. A Tritium Extraction System (TES) has been also integrated in the 2013 Blanket CAD model. The connection with the Back Supporting Structure has been realized with the integration of the new box module geometry and also the dual cooling layout in a compact solution, which eliminates any gap between the modules back plate and the BSS. The attachment between the Vacuum Vessel and the BSS has been realized taking into account the mechanical, thermal and electromagnetic loads linked to the new design. The paper presents the status of the HCPB DEMO Blanket design, also describing the results of the performed analyses to underpin the proposed solution.

Id 395

Abstract Final Nr. P2.122

Optimization of V-Shaped one-side-Ribbed Channel for Helium Cooled DEMO First-Wall

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Helium gas as a coolant offers several advantages in terms of safety. However due to the less favourable thermophysical properties compared to liquid coolants, the use of standard smooth cooling channel surfaces are limited regarding very high heat flux densities from the plasma in the range of 1MW/m² predicted for some fusion power reactor scenarios. Based on our previous assessments, a round-edged, one-side-ribbed rectangular channel was chosen as the baseline geometry, with the ribbed-side facing the plasma-facing wall. The V-shaped rib with an angle of attack of 120° was adopted, which was optimized for manufacturing by milling; its cross section is not square but trapezoidal. This paper presents the optimization of the V-shaped ribbed channel by means of CFD simulations. The basic principle is to promote the turbulent mixing and extend the heat transfer area. Thus the effects of the rib pitch and the employ of various additional ribs (turbulence promoters) are studied. The main criteria for assessing the results are the heat transfer coefficient which determines the maximum wall temperature and, the friction factor which determines the pumping power. The results are also compared with the standard transversal ribbed channel and smooth channel.

Id 711

Abstract Final Nr. P2.123

Modelling and Shielding Analysis of the Neutral Beam Injector Ports in ITER

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The ports for the neutral beam injectors (NBI) in ITER represent openings to the plasma chamber which lead to an intense neutron radiation streaming along the NBI ducts. Neighbouring components such as the vacuum vessel (VV), the NBI duct liners and the toroidal field coil (TFC) magnets are most exposed to this radiation. It is thus required to proof that the assumed shielding around the NBI ports is sufficient for the protection of these components from the impinging intense radiation. To this end a suitable neutronic model of the ITER NBI sector need to be prepared and radiation transport calculations need to be performed to assess the radiation loads to the VV and the TFC in the vicinity of the NBI ducts. In this work, a new MCNP geometry model of the NBI ports was developed starting from the latest engineering CAD models provided by ITER. The model includes 3 heating (HNBI) ports and the diagnostic port (DNBI) and extends up to the bio-shield. The engineering CAD models were simplified on the CATIA platform according to the neutronic requirements and then converted into MCNP geometry making use of KIT's McCad geometry conversion tool. Finally, the new NBI port model was integrated into an available 80o ITER torus sector model. The nuclear analysis performed on this model provides the following nuclear responses: the neutron flux distribution in all NBI ports, the nuclear heating distribution in the VV and all NBI ducts; the nuclear heating and radiation loads to the TFC magnets; the radiation damage and gas production in the VV; and the distitribution of the shutdown dose rate inside the cryostat. The particle transport simulations were performed using the MCNP5-1.6 code and the FENDL-2.1 data library. Results were generated on high resolution mesh grids making use of MCNP's mesh tally feature. KIT's recent mesh based rigorous two step approach (R2Smesh-2.1) was applied for the calculation of the shutdown dose rate distributions. The Paraview software was used to visualize the mesh distribiutions on the underlying CAD geometry.

Id 417

Abstract Final Nr. P2.124

Creep fatigue assesment macro in MAPDL for EUROFER

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Feature of high dose irradiation resistance and creep strength at high temperatures makes EUROFER a strong candidate for in-vessel components of fusion reactors. This material in blanket and divertor is subjected to high cyclic thermo-mechanical loads yielding creep-fatigue. For reliable evaluation of fusion reactor system analysis of failure among others due to creep-fatigue interaction is essential for the engineering design. Therefore investigation of time dependent creep-fatigue at high temperature is one of the major issues in the development of fusion reactors. In the frame of Engineering Data & Design Integration (EDDI) in EUROFUSION Technology Work Programme fast design assessment is required for in-vessel structures built from EUROFER to predict the critical damaged regions under typical fusion reactor loads. Within this work creep-fatigue damage is evaluated using the elastic analysis approach of the American Society of Mechanical Engineers - Boiler Pressure Vessel Code (ASME-BPVC) and adopting it for EUROFER. Required design fatigue and stress-to-rupture curves for estimating creep-fatigue damage are collected from published EUROFER data evaluations. Loading parameters, e.g. local stress, maximum elastic strain range and temperature, needed for the analysis of creep-fatigue damage are delivered from Mechanical ANSYS Parametric Design Language (MAPDL). To investigate the influence of pulsed mode with different pulse durations (hold-times) on creep-fatigue damage a FORTRAN code for creep fatigue assessment (CFA) has been developed as a post processor coupled with MAPDL. This code was only able to analyze creep fatigue damage on predefined paths at different pulse durations. Within the work to be presented here the CFA code has been further developed to consider irradiation influence and to identify automatically the most critical region of analyzed component. The modified CFA code has been applied to a preliminary design of DEMO blanket demonstrating its capability for the creep-fatigue lifetime assessment of geometrically arbitrary shaped components under fusion reactor conditions.

Id 746

Abstract Final Nr. P2.125

Development of a zonal applicability tool for remote handling equipment in demo

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A radiation-induced damage caused by the ionizing radiation can induce a malfunctioning of the Remote Handling Equipment (RHE) in fusion reactors, nuclear power stations and high-energy accelerators. Therefore to achieve a sufficient length of operational time inside future fusion power plants, a suitable radiation tolerant RHE for maintenance operations in radiation environments is inevitably required. Thus to assess the influence of the radiation (this work is exclusively concentrated on the damages created by high energetic gamma rays), an investigation about radiation hardness assessment and possible failure modes of various components that are typically used in RHE was performed. Additionally information about the total dose rate that every component can withstand before failure was collected. Furthermore a zonal applicability tool was developed using Excel VBA to help RHE designers. The tool connects the data from the radiation filed analysis (3-D radiation map) to the radiation hardness data of the planned RHE for DEMO maintenance. The intelligent combination of the available information (radiation behavior and radiation level at certain time and certain location) will allow to take decisions regarding the application of specific RHE. The user inputs the following parameters: the specific device used in the RHE, the planned location (inside or outside the vacuum vessel) and the maintenance period (1 week, 1 month, 1 year after shutdown). The output is the expected lifetime of the selected RH component at the given location and maintenance period. Planned action times have to be also considered (data for the planned action times of RHE should be added in the future). After having all the parameters it can be decided, if dedicated RHE will survive the duration of the maintenance operation.

Id 858

Abstract Final Nr. P2.127

A coupled systems code - CFD MHD solver for fusion blanket design

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Fusion blankets are required to operate in a harsh environment under the influence of a number of interdependent and synergistic physical phenomena, working across several length scales. For magnetic confinement reactor designs using a conducting fluid as coolant/breeder, the difficulties in flow modelling are particularly challenging due to interactions with the large magnetic field. Despite the numerous design concepts that have been proposed, blankets comprise a number of common features such as ducts, manifolds and connections. Ducts are amenable to simplified 1D treatment via a thermal-hydraulics systems code, enabling the efficient, agile simulation of the blanket at a component level. In more complex components such as manifolds and junctions, the flow is truly 3D in nature, and CFD has traditionally been used for such analysis. However, as a result of the exorbitant run times accurate models are commonly limited to simple 2D geometries at relatively low Hartmann numbers. Blankets are therefore an ideal candidate for the application of a code coupling methodology, with a thermal hydraulic systems code modelling portions of the blanket amenable to 1D analysis, and CFD providing detail where necessary. It is the aim of this study to develop such a modelling approach, enabling extensive thermal hydraulic simulation of the blanket and associated systems and accounting for MHD effects in a computationally efficient manner that lends itself to the design process. A systems code with MHD capability has been developed and validated against existing analyses. The code shows good agreement in the prediction of MHD pressure loss and the temperature profile in the fluid and wall regions of the blanket breeding zone. The systems code has been coupled to an MHD solver developed in OpenFOAM, via TCP socket connections. This coupled solver has been validated for several geometries in preparation for modelling extensive blanket systems.

Id 957

Abstract Final Nr. P2.128

Re-Assessment of Tritium Self-Sufficiency for Fusion Reactor by Dynamic Model

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Tritium is important fusion fuel and there is no practical external source for fusion energy. The fusion reactors should breed their own tritium in blanket to support the plasma burning in vaccuum vessel. The International Thermonuclear Experimental Reactor (ITER) will be the first fusion device for Deuterium-Tritium plasma burning, but without full fuel cycle of tritium. To demonstrate tritium self-sufficiency and other key issues, an engineering test reactor is necessary to be constructed parallel with ITER and before Demo reactor, such as China Fusion Engineering Testing Reactor (CFETR) with fusion power 50-200MW, duty cycle 0.3~0.5. In this work, a dynamic fuel cycle model had been developed by using system dynamics method for tritium self-sustaining analysis, management, and safety analysis. In the upgraded model, the fuel circulation scheme was optimized, both pulse and steady-state operations were coverd, and impact of duty time was intruduced. To validate the accuracy an effectiveness of this model, a benchmak of typical fusion test reactor had been descripted in detail, extrapolated from ITER device. And the key effect factors of tritium self-sufficiency were taken into analysis and discussion in detail, including the operation mode and availability, tritium burn up fraction in plasma, and processing capacity of tritium system. After the sensitive analysis, the initial inventory required for startup and minimum TBR required for self-sufficiency were given basing on the consideration of state-of-the-art. Finally, compared with the achievable TBR of blanket, the required conditions of D-T burning plasma, tritium technology and system reliability to ensure tritium self-sufficiency were indicated.

Id 571

Abstract Final Nr. P2.129

Analysis on the Testing Results of MHD Effect and Related Measuring Accuracy in DRAGON-IV PbLi Loop

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Dual Functional Lead Lithium (PbLi) test blanket module (DFLL-TBM) is considered as one of the TBM candidates for ITER, PbLi is designed as tritium breeder, neutron multiplier and coolant. Since the conductivity of PbLi itself and strong magnetic field around TBM, the magnetohydraudynamic (MHD) effect will affect the characteristic of PbLi flow field compared to the case without magnetic field. Therefore, how to get the real distribution of local PbLi flow field, especially for pressure and velocity, under magnetic field becomes one of the key issues of the liquid metal blanket and with less experiment results up to now. A multi-functional PbLi loop DRAGON-IV was built to study the main testing objective of MHD effect, measuring technologies of flow velocity and pressure drop inside PbLi flowing channels compared to the theory calculation results. Because of the requirement of high accuracy for meters to ensure the reasonability of testing results, it is important to take more attention for the assembly of pressure differential meters in the test section, and make the two connection points in the flow channel keeping at the same level which could reduce the results error. As to the flow meter with no international common standard of measurement accuracy, the calibration process was proposed and the data of flow rate was obtained by every two level meters which were recorded by computer automatically. Following the test strategy of China PbLi blanket, the first test was finished with low testing condition that the magnetic field is 1.8 tesla, flow rate is 0.1m/s and temperature is 350º, the test results were compared to the values calculated by MTC code with the error of less 10%, which will support greatly to the design and engineering technologies of China PbLi blanket. Keywords: PbLi blanket; DRAGON-IV; MHD effect; Measuring accuracy

Id 906

Abstract Final Nr. P2.130

Simulations of Effect of Off-Normal Events on LLCB TBM First Wall

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The First Wall (FW) is one of the most important components of any fusion blanket design. Indian Test Blanket Module (TBM) program in ITER is one of the major steps in its fusion reactor program towards DEMO and future Fusion Power Reactor (FPR) vision. India has proposed Lead–Lithium Cooled Ceramic Breeder (LLCB) as the blanket concept to be tested from the first phase of ITER operation in one-half of an ITER port no. 2. FW of LLCB TBM is having Reduced Activation Ferritic Martensitic Steel (RAFMS) as the structural material actively cooled by high pressure Helium gas. Cooling channels are running in radial–toroidal–radial direction in the RAFMS structure. The FW is designed to withstand the withstand the maximum coolant pressure of 8 MPa, energetic particle fluxes and heat fluxes from the plasma, high thermal and mechanical stresses and magnetic forces during plasma disruptions and other ITER transients. On top of the steady state heat loads, during transient events a large amount of energy with densities up to several MJ/m2 is deposited in extremely short time periods on the FW of ITER plasma facing components including TBMs. In addition to erosion and melting, severe damage of the FW and coolant channels may occur during these transients. Thermal–hydraulic modeling and simulations have been performed for different ITER transient thermal loads such as plasma disruptions, Edge Localized Modes (ELMs) and MARFE in terms of power density and pulse duration to ensure the proper performance of FW of LLCB TBM. The detailed analysis and optimization of their performance will be discussed in this paper.

Id 799

Abstract Final Nr. P2.131

Blower Gun pellet injection system for W7-X

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Foreseen to perform pellet investigations in the new stellarator W7-X,the former ASDEX Upgrade Blower Gun was revised and revitalized. The systems operational parameters have been characterized in a test bed. The gun is capable to launch cylindrical pellets with 2 mm diameter and 2 mm length, produced from frozen Deuterium (D2), Hydrogen (H2) or a gas mixture consisting of 50% H2 and 50% D2. Pellets are accelerated by a short pulse of pressurized helium propellant gas to velocities in the range of 100-250 m/s. Delivery reliabilities at the launcher exit reach almost unity. The initial pellet mass is reduced to about 50% during the acceleration process. Pellet transfer to the plasma vessel was investigated by a first mock up guiding tube version. Transfer through this S-shaped (inner diameter 8 mm; length 6 m) stainless steel guiding tube containing two 1 mcurvature radii was investigated for allpellet types. Tests were performed applying repetition rates from 2 Hz to 50 Hz and propellant gas pressures ranging from 1 to 6 bar. For both H2and D2, low overall delivery efficiencies were observed at slow repetition rates, but stable efficiencies of about 90% above 10 Hz. About 10% of the mass is eroded while flying through the guiding tube. Pellets exit the guiding tube with an angular spread of less than 14°.

Id 390

Abstract Final Nr. P2.132

Preparation of Acceptance Tests and Criteria for the Test Blanket Systems to be operated in ITER

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ITER provides a unique opportunity to test tritium-breeding technologies needed for the development of fusion as an energy source. Various mock-ups of breeding blanket concepts, currently developed by ITER Members (IM), will be tested in ITER Test Blanket Modules (TBM) Program. The target for breeding blanket development is demonstration of tritium self-sufficiency and extraction of high grade heat and electricity production in a reactor environment. The TBM program foresees operation of six different Test Blanket Systems (TBS), each one including a TBM involving either flowing Pb-16Li eutectic alloy and/or ceramic pebbles of ternary lithium ceramics as breeding material. Coolants used are helium, water, and/or Pb-16Li. The TBSs are designed to very high quality standards to meet the nuclear safety requirements and high level operational targets set for ITER. The design and fabrication of the TBSs are performed by IM's that will deliver them to the ITER site.. Integration of these components within the ITER Vacuum Vessel and Tokamak Complex building will be performed by the IO, with the support of the IM. Each TBS is formed by about 70 to 100 major components, depending on the TBS type. Different components have different classifications (e.g. Safety, ESP/ESPN, Seismic, Quality, Tritium, Vacuum, Remote Handling). The paper elaborates on identification of the acceptance tests and criteria, that: • need to be performed in the factory; • need to be performed on each of the TBS components (or type of components) after their delivery to the ITER Site; • can only be performed on systems, and those during TBS commissioning (i.e. after TBS installation); • are foreseen during the Long Term Maintenance State (LTM) before restarting the Plasma Operation State (POS). This listing is not necessarily exhaustive, and complementary to the acceptance tests already required by the applicable codes & standards, and French regulations (depending on the components classifications).

Id 156

Abstract Final Nr. P2.133

Development of the ITER roughing pumping system

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The ITER vacuum system will be one of the largest, most complex vacuum systems ever to be built. There are a number of large volume systems including: the Cryostat (~ 8500m3), the Torus (~1330 m3), the Neutral Beam injectors (~180m3 each) and a number of lower volume systems including: the service vacuum system, diagnostic systems, and electron cyclotron transmission lines. The roughing pumping system is required to initially evacuate these systems, to continuously back certain types of high vacuum pumps and to pump regeneration gas from large cryopumps. Consideration for safe handling of the process gas are key throughout the systems design. Particularly challenging requirements exist on the roughing system for both, confinement and pumping performance as a result of the high hydrogen isotope throughputs and the variety of other gases in the tokamak exhaust. In combination with adapted, conventional mechanical pump technology a concept which utilizes a novel pump named a Cryogenic Viscous flow Compressor (CVC) is under development to meet the pumping requirements. The principle of CVC is that it will cryogenically condense hydrogen isotope mixtures, originating from the Torus and Neutral Beam cryopumps, while providing first stage compression of helium ash originating from the plasma burn. The development program for the CVC first involved extensive modeling, second a small scale prototype tested using a gaseous cryogenic helium supply, and thirdly a full scale prototype which is being constructed for testing using a super critical helium supply, similar to that which will be used at ITER. In this paper an overview of the overall design and development status of the roughing systems is given and specifically the results of tests with deuterium and helium on the representative small scale prototype CVC are evaluated. Details are given on how these results affect the next development and design stages.

Id 708

Abstract Final Nr. P2.134

Progress of Design and Fabrication Technology Development of ITER Test Blanket Module of Water Cooled Ceramic Breeder Blanket in JAEA

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Japan Atomic Energy Agency (JAEA) is performing the design and fabrication technology development of ITER Test Blanket Module (TBM) of Water Cooled Ceramic Breeder (WCCB) Blanket as one of the most important steps toward DEMO blanket. Regarding the blanket module fabrication technology development using F82H, the fabrication of a real scale mockup of the TBM box was performed. In support to the integrity confirmation on corrosion of structural material, the basic data on flow assisted corrosion of F82H in high pressure and temperature water was performed. In the design activity of the ITER TBM, the structure design of the shield and electromagnetic analysis under plasma disruption events were performed. Safety analysis of the TBM and the auxiliary system was performed using TRAC code. As for the design of the auxiliary system, conceptual design was performed on the tritium extraction system and water cooling system. In support to the design of the neutronics measurement system, preliminary investigation of the measurement feasibility of various method was performed. This paper overviews the recent progress of the design and fabrication technology development of ITER TBM of WCCB Blanket in JAEA.

Id 159

Abstract Final Nr. P2.135

Upgrade in catalytic activity of hydrophobic platinum catalysts by irradiation with electron beams

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Hydrophobic platinum catalysts have been widely applied in the field of nuclear fusion for the exchange reactions of hydrogen isotopes between hydrogen and vapor in the water detritiation system, and for the oxidation of tritium on the atmospheric detritiation system. Hydrophobic platinum catalysts are hardly susceptible to water mist and water vapor. Hydrophobic platinum catalysts are produced by supporting platinum directly on hydrophobic polymer beads. For the hydrophobic polymer, styrene - divinyl benzene (SDB) has been applied in Japan. It can be pointed out that the upgrade in catalytic activity of hydrophobic catalyst is expected to downsize the catalytic reactor based on a hard look at a large increase in flow rate in future. The upgrade in catalytic activity of two types of commercial Pt/SDB catalysts was found when they were irradiated with electron beams. After irradiation with electron beams, the catalytic activity was evaluated by means of overall reaction rate constant for the oxidation of tritium. The overall reaction rate constant increased as increase in dose. The constant showed the peak value in the dose between 500 to 750 kGy. After the peak, the constant decreased as increase in dose. The overall reaction rate constant at the peak was 6 times larger than that evaluated with unirradiated. The mechanical strength of irradiated Pt/SDB kept sound until 1500 kGy. The irradiation is a promising method to the upgrading in catalytic activity of Pt/SDB catalyst.

Id 190

Abstract Final Nr. P2.136

Fabrication and hydrogen generation reaction with water vapor of prototypic pebbles of binary beryllides as advanced neutron multiplier

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Advanced neutron multipliers with high stability at high temperatures are desired for the pebble bed blankets of DEMO reactors. Beryllium intermetallic compounds (beryllides) are the most promising material for this purpose. To fabricate the beryllide pebbles, a new granulation process has been established that combines a plasma sintering method for beryllide synthesis and a rotating electrode method using a plasma-sintered electrode for granulation. In granulation examinations, prototypic pebbles 1 mm in diameter of Be-V beryllide as well as Be-Ti beryllide were successfully fabricated. This study performed not only granulation of binary beryllides but also its characterization of the hydrogen generation reaction with water vapor compared with those of pure Be pebbles. The reactivity test at 1273 K for 24 h under Ar-1%H2O atmosphere revealed that amount of hydrogen generation of these beryllides pebbles was lower by about two orders of magnitude than that of Be pebbles. These experiments also showed that the weight gain ratios of these beryllides pebbles were significantly smaller than those of Be pebbles. Be pebbles swelled with oxidation and exhibited many cracks on the surface of the oxidation layer. However, BeO deposited to over the entire surface of these beryllides pebbles, and there were no swelling and no cracking. During oxidation of Be, compressive stress induces cracks within the Be scale. This is because the lattice coherency between the substrate and the scale is destroyed. On the other hand, during oxidation of these beryllides, stress-free BeO can be formed on the surface of these pebbles because the atomic distances between Be atoms within the BeO scale are similar to the mean atomic distance between Be atoms within these beryllides substrates. The results confirmed that prototypic pebbles of Be-Ti and Be-V beryllides have better reactivity resistance with water vapor than pure Be pebbles.

Id 204

Abstract Final Nr. P2.137

Integral Test of International Reactor Dosimetry and Fusion File on Graphite Assembly with DT Neutron at JAEA/FNS

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A new library of dosimetry cross sections, International Reactor Dosimetry and Fusion File release 1.0 (IRDFF 1.0), has been released from the International Atomic Energy Agency (IAEA) recently. The energy limit of the cross section data in IRDFF 1.0 is extended to at least 60 MeV from 20 MeV in the former International Reactor Dosimetry File 2002 (IRDF-2002) for fusion applications in addition to reactor dosimetry. Although the reaction data in IRDFF 1.0 are derived partly from those in IRDF-2002, about a half of those are newly evaluated. In order to validate and test IRDFF 1.0, IAEA has initiated a new Co-ordinated Research Project (CRP). Under this CRP, we have performed an integral experiment on a graphite pseudo-cylindrical slab assembly with DT neutron source at JAEA/FNS. The graphite assembly of 31.4 cm in equivalent radius and 61 cm in thickness is placed at a distance of about 20 cm from the DT neutron source. A lot of foils for the dosimetry reactions in IRDFF1.0 are inserted into the small spaces between the graphite blocks along the center axis of the assembly. After DT neutron irradiation, reaction rates for the dosimetry reactions are measured by the foil activation technique. This experiment is analyzed by using Monte Carlo neutron transport code MCNP5-1.40 with recent nuclear data libraries of ENDF/B-VII.1, JEFF-3.2, and JENDL-4.0. The experimental assembly and DT neutron source are modeled precisely in the MCNP calculation. The reaction rates calculated with IRDFF 1.0 as the response functions for the dosimetry reactions are compared with the experimental values. Also the calculations with JENDL Dosimetry File 99 (JENDL/D-99) are performed for comparison. The results calculated with IRDFF 1.0 show good agreement with the experimental results.

Id 506

Abstract Final Nr. P2.138

Effect of beryllium contents in titanium beryllide pebbles on crush strength and oxidation resistance

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Titanium beryllium intermetallic compounds, beryllides, have been investigated as a candidate of advanced neutron multipliers in water-cooled solid breeder concept in a fusion demonstration reactor, under researching on the broader approach (BA) activities at the International Fusion Energy Research Centre (IFERC) in a joint Japan/EU project. We have suggested a combinational process for the beryllide pebble fabrication, consisting of plasma-sintering method and rotating electrode method for fabrication of electrode rods and pebbles, respectively. To investigate the effect of Be contents in titanium beryllide pebbles on the crush strength and oxidation resistance, the beryllide pebbles with 3 to 10.5 at.% Ti were fabricated. As a result of SEM observation, it was clear that Be phase on the surface identified in Be- 3 and 5 at. % Ti while no Be phase found in Be-7 to 10.5 at.% Ti. According to cross-sectional images, moreover, area fraction of Be phase decreased with decrease of Be content in 3 to 9 at. % Ti beryllide pebbles whereas the pebble with 10.5 at.% Ti was mainly consisted of $Be_{17}Ti_2$ phase. It was confirmed from crushed tests of the pebbles that ductility increases as increased of Be contents in the pebbles since the existence of Be phase leads to ductile fracture. On the other hand, Be phase on the surface of the pebbles with Be- 3 and 5 at. % Ti resulted in increase of the weight gain due to oxidation while other pebbles indicated similar values because the surface of pebbles with 7 to 10.5 at.% Ti consisted of either $Be_{12}Ti$ or $Be_{17}Ti_2$. Accordingly, this existence of Be phase in the pebble leads to increase of ductility while it results in decrease of oxidation resistance on the surface at 1273 K.

Id 499

Abstract Final Nr. P2.139

Experimental Investigation on Tritium Recovery from Lithium Titanate Pebble under high temperature of 1073K

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In fusion DEMO reactors, it is assumed that the inside temperature of Li2TiO3 pebble breeder blanket is more than 1000 K. For the investigation of tritium recovery on a Li2TiO3 pebble breeder blanket, we have carried out tritium recovery experiment at 1073 K with an external heating system and DT neutron source at the JAEA-FNS. The Li2TiO3 pebble of 70 g was put into a stainless steel container and it was installed into an assembly stratified with beryllium and Li2TiO3 layers. The Li2TiO3 pebble was heated at 1073K with a heating wire during DT neutron irradiation. Two air compressors were simultaneously run to increase the coolant capacity during the DT neutron irradiation. Helium gas including 1 % hydrogen gas (H2/He) was mainly flowed in the container inside. Two chemical forms, HT and HTO, of extracted tritium were separately collected during DT neutron irradiation by using water bubblers and CuO bed. The tritium activity in the water bubbler was measured by a liquid scintillation counter. Moisture (H2O) in the gas was suppressed below 100 ppm with a molecular sieve. In contract, to investigate the effect of moisture in the sweep gas, we also performed the same experiment with H2O/He gas (H2O content : 1%). From the present experiment, it was shown that the HT gas extracted at 1073 K was more than 90% of collected tritium in the case of the H2/He sweep gas. However, that of the H2O/He sweep gas was 50%. In order to collect produced tritium as HT, it is essential to reduce the water content in the purge gas.

Id 519

Abstract Final Nr. P2.140

Optimization of Granulation Conditions of Advanced Tritium Breeder Pebbles using the Emulsion Method

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Demonstration power reactors (DEMOs) require advanced tritium breeders that have high stability at high temperatures. Lithium titanate (Li_2TiO_3) is one of the most promising candidates among tritium breeders. However, a decrease in lithium mass of Li_2TiO_3 with time occurs in such environments as the DEMO blanket because of Li evaporation and Li burn-up. Therefore, an original material of Li_2TiO_3 with excess Li ($Li_{2+x}TiO_{3+y}$) as an advanced tritium breeder that can make up to the lithium loss has been proposed. Pebble fabrication using the emulsion method is one of the promising techniques for the mass production of the advanced tritium breeder pebbles. The authors have been developing a technique of fabricating $Li_{2+x}TiO_{3+y}$ pebbles using the emulsion method. In a previous study, The average grain size on the surface and the cross section of the sintered $Li_{2+x}TiO_{3+y}$ pebbles was < 5 µm and 5 – 10 µm, respectively. Considering the tritium release characteristics and the packing factor of the blanket, the desired pebble diameter and grain size after sintering were 1 mm and < 5 µm, respectively. Therefore, the next step was to optimize the granulation conditions to reach these target values. The grain growth factor is assumed to be the presence of binder in the gel particles. This remaining binder reacts with the excess Li in the $Li_{2+x}TiO_{3+y}$, and Li_2CO_3 is generated. To prevent the generation of Li_2CO_3, calcined $Li_{2+x}TiO_{3+y}$ pebbles were sintered in a vacuum atmosphere at 1073 K for 3 h and in a 1%H_2-He atmoshere at 1323K for 5 h. The average grain size on the surfaces and cross sections of the sintered $Li_{2+x}TiO_{3+y}$ pebbles was < 5 µm. In addition, the diameter of sintered $Li_{2+x}TiO_{3+y}$ pebbles was 1.07mm. These results show that the $Li_{2+x}TiO_{3+y}$ pebbles were successfully fabricated by using the emulsion method.

Id 678

Abstract Final Nr. P2.141

Improvement of CAD/MCNP conversion system GEOMIT

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By performing the nuclear analyses with the Monte Carlo code MCNP and the detailed calculation geometry data, nuclear properties such as neutron fluxes, nuclear heating, dose rate, etc., can be obtained with high accuracy in nuclear design of nuclear fusion reactors. It takes much huge time to manually create calculation geometry input data for complicated structure such as fusion reactors, which is a critical concern. In order to solve this concern, we have been developing an automatic conversion system, GEOMIT, from CAD data to MCNP geometry input data. MCNP requires an explicit definition for the void data, which are not expressed in CAD data at all. Thus the conversion system should be able to create the void data and to convert the void and solid in the CAD data to MCNP geometry input data. Now GEOMIT is greatly improved in this study, though it had many problems in the previous version. Main improvements are as follows. (1) The methodology is revised on the generation of the ambiguous surface and the function of the surface reduction. In case one surface fits other surfaces within the predefined tolerance, the function can merge those surfaces into one surface. (2) The conversions of the solid and void data were performed with the different processes in the previous version. They can be performed with just one process in this version. (3) Boxes for void data are newly created in GEOMIT. These boxes can be automatically divided into ones with appropriate size in this version. Conversion results are improved by these revisions, and the MCNP geometry errors disappear. The CAD data of the JA WCCB ITER/TBM is converted to the MCNP geometry data by the revised GEOMIT, and it is confirmed the conversion can be successfully performed.

Id 699

Abstract Final Nr. P2.142

R&D activities of tritium technologies on Broader Approach in Phase 2-2

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Activities on Broader Approach (BA) were started in 2007 on the basis of the Agreement between the Government of Japan and the EURATOM. The period of BA activities consist of Phase1 and Phase2 dividing into Phase 2-1 (2010-2011), Phase 2-2 (2012-2013) and Phase 2-3 (2014-2016). Tritium technology was chosen as one of important R&D issues to develop DEMO plant. R&D activities of tritium technology on BA consist of four tasks. Task-1 is to prepare and maintain the tritium handling facility in Rokkasho BA site in Japan. Task 2, 3 and 4 are main R&D activities for tritium and these are focused on: Task-2) Development of tritium accountancy technology Task-3) Development of basic tritium safety research Task-4) Tritium durability test At the R&D activities in Phase 1, the collaboration research programs between Japan Atomic Energy Agency (JAEA) and Japanese universities were started and many studies have been carried out by JAEA as well as Universities. Presentation give details of the progress of R&D activities of tritium technology in Phase 2-2 together with overlook of significant results obtained on above-mentioned tasks and the plan of R&D activities in Phase 2-3 is also presented.

Id 856

Abstract Final Nr. P2.143

Design study of blanket structure based on a water-cooled solid breeder for DEMO

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Blanket concept with simplified interior for mass production has been developed with a mixed bed of Li2TiO3 and Be12Ti pebbles, a coolant condition of 15.5MPa and 290-325ºC and cooling tubes only without any partitions. A neutronics analysis ensured the blanket concept meets a self-sufficient supply of tritium. However, this concept is vulnerable to the inner pressure. A plant availability for DEMO may drop to a lower value, because a potential of resume operations after an accident such as a coolant leakage in blanket is not considered. The blanket design will be revisited for the availability. Considering the continuity with the ITER-TBM option of Japan and the engineering feasibility of fabrication, our design study focuses on a water-cooled solid breeding blanket using the mixed pebbles bed. A breakage of the blanket casing should be avoided not to contaminate the plasma chamber with water and breeding materials. A water-cooled solid blanket with inner pressure tightness is estimated by the ANSYS code. As a results, the pressure tightness of 8MPa (water vapor pressure at 300ºC) can be compatible with the self-sufficient production of tritium when the blanket is as thick as about 0.9m and the ribs are arranged in the radial direction. Therefore, the blanket concept with pressure tightness of 8MPa is adopted with depressurization system as which a tritium recovery system such as helium purge-gas line is posteriorly arranged in blanket to serve. On the other hand, a handling of decay heat is a serious problem at an accident such as LOCA. Coolant flow is divided into the blanket to secure heat removal for the safety. Finally, the blanket segmentation with the shape and dimension of blanket and routing of coolant flow has also been proposed. Moreover, overall TBR is estimated with torus configuration based in the segmentation using three-dimensional MCNP calculation.

Id 953

Abstract Final Nr. P2.144

Development of fabrication procedure for Korean HCCR TBM sub-module using ARAA

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Korea has developed and plans to test a Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM) in the ITER. The HCCR TBM is composed of four sub-modules and a back manipulator (BM). Each sub-module is composed of a first wall (FW), breeding box, and side walls (SW). The front surface of the sub-module is 231 mm in width and 835 mm in height. In the FW of the sub-module, there are 11 rectangular shaped cooling channels with 15 mm in width and 11 mm in height. The fabrication method of the breeding zone of the HCCR TBM was newly designed. The fabrication procedure was developed to confirm the fabrication method for the HCCR TBM. As the HCCR TBM structural material, Advanced Reduced Activation Alloy (ARAA) is being developed. The first 5 ton commercial-scale ARAA material was fabricated and called ARAA-1. The test specimens of the ARAA-1 were prepared to test the weldability for tungsten inert gas (TIG) welding and electron beam (EB) welding. To establish and optimize the welding procedure in an EB weld from ARRA-1 material, the variation in the bead width and penetration depth according to the welding current and welding speed were investigated. To verify the welding performance of newly developed ARAA-1 steel for TIG welding and E-beam welding, a series of tensile, face bending, root bending, and V-notch impact tests were carried out.

Id 226

Abstract Final Nr. P2.145

Design and construction of a helium cooling system for the HCCR TBM

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A real-scale helium supplying system (HeSS-RS) has been designed and constructed at Korea atomic energy research institute (KAERI) to demonstrate the helium cooling system (HCS) of helium cooled ceramic reflector test blanket module (HCCR TBM) for ITER project and to obtain thermal-hydraulic experimental data for codes validation. The loop, HeSS-RS, is designed to operate at high temperature (up to 500º) and high pressure (up to 10 MPa) conditions with helium mass flow rate of 1.5 kg/s. A circulator, which is one of the essential components of HeSS-RS, has been specially developted to provide helium compression rate up to 1.1 during normal operation (300º, 8 MPa, 1.1 kg/s of helium flow). The normal operation of HeSS-RT is simulated using the GAMMA-FR code to design the HeSS-RS system and the analysis result is compared with the HeSS-RT test run data for code validation.

Id 834

Abstract Final Nr. P2.146

Integrated Design and Performance Analysis of the KO HCCR TBM for ITER

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To develop a Fusion Reactor, we have participated in the TBM program in the ITER. Based on the separate analysis with the functional components such as FW, BZ, SW, and BM by 2012, an integrated analysis model was prepared and preliminary analysis was performed. Considering the flow manifold like SW and BM, more actual flow distributions were obtained and it is confirmed that the structure temperatures were different with the old analysis considering the uniform flow rates. Some regions exceeds the design limit (550 oC) and design optimization with SW and BM will be performed. After the thermal-hydraulic optimization, structural analysis with ANSYS-mechanical will be performed considering the internal pressure of 10 MPa design pressure.

Id 463

Abstract Final Nr. P2.147

Internal advection mechanism of a falling liquid Pb-17Li droplet for hydrogen isotopes recovery

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The authors reported the mass transport enhancement of hydrogen isotopes on the liquid Pb-17Li droplet, falling in vacuum, and proposed its application for tritium recovery from LiPb blankets. This study aimed to identify the enhancement mechanism of a falling droplet. The spherical oscillation of a droplet and corresponding internal advection was theoretically analyzed. By calculation, the basic eigen frequeny of the droplet is as the function of diameter, and deforms between a prolate and oblate spheroid. The corresponding flow is a poloidal reciprocating movement between the north pole or south pole and the equator. By this current, the disolved hydrogen isoptes are advected to the surface drastically faster than by the diffusion mechanism. Through the fresh surface, which is emerged by the cyclic deformation of a shpere, the hydrogen isotopes can be released without being limited by the effect of 2nd order recombination rate. The feasibility of internal vortex motion was also analyzed. As the droplets are falling in vacuum, the shear stress to generate vortex flow can not exist. It is posssible only at the detachement of dripping condition, which is available only at the velocity lower than the operating condition. The experiment of falling droplets of PbLi was performed by a high speed movie with 0.6mm nozzle 2.5 m/s velocity condition, and the oscillation was observed. All droplets showed the basic mode of spherical harmonics oscillation, as predicted. This oscillation is induced by the detached ligaments. It is considered that the ligaments repelled toward the core of each droplet at the detach, but did not induce the internal vortex flow. The density of PbLi and the detachment frequency are considered to contribute this result. The result suggests that the internal advection of a falling droplet by an oscillation as a cause of the mass transport enhancement.

Id 762

Abstract Final Nr. P2.148

A system dynamics model for stock and flow of tritium in fusion power plant

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The flow and stocks of tritium as a fuel of D-T fusion power plants will be affected by many variables in many primary fuel cycle components such as plasma (PL), fuel clean up system (FCU), isotope separation system (ISS), fueling and storage system (FS), breeding blanket system (BLK) and recovery from gaseous, liquid and solid wastes as secondary. Therefore a designer integrating the total system needs to share information with the other specific researchers and engieers. System dynamics (SD) models are useful to visualize and analyze tritium fuel system. In this study, a SD model has been developed using a commercial software STELLA. The five components (PL, FCU, ISS, FS, BLK) are described as main stocks for tritium. In the main flow, tritium flow at a stock Nj from upstream stock Ni and to downstream stock is defined with their mean residential times πi and πj as dNj / dt=Ni / πi – Njπj. In addition, prodcution and burning through fusion reaction (D-D and D-T) and breeding of tritium at blanket are also considered in the model. Decay and leak of tritium going to the secondary systems can be also defined. Inactive inventory that exists in the primary components with much longer time constant is also expressed as leak term. By using this model, we surveyed a possible parameter window for D-D start-up without initial loading of tritium, suggested by Konishi et al.. Beyond ITER, acquiring tritium may become an obstacle to initiate DEMO program in Japan because of no available commercial tritium. The D-D start-up scenario can reduce the necessity of initial loading of tritium through the production in plasma by D-D reaction and in breeding blanket by D-D neutron. The model was also used for analyzing operation scenario of DEMO to prevent excess stock of tritium which must be produced at tritium-breeding-ratio over unity.

Id 763

Abstract Final Nr. P2.149

Lithium desorption capacity from Li2TiO3 and Li4SiO4

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Water vapor is released from ceramic breeder materials into the purge gas due to desorption of adsorbed water and water formation reaction. The released water vapor possibly promotes Li mass loss with the formation of LiOH on the surface. In fact it has been observed by the present authors that the amount of Li mass loss under 50Pa H2O/Ar is larger than that under 1000Pa H2/Ar. In this study, Li mass loss from Li2TiO3 and Li4SiO4 was investigated focusing on the relationship of Li desorption and water vapor desorption. Sample pebbles used in this study were Li2TiO3 with excess Li (by JAEA (Li/Ti=2.11)) and Li4SiO4 (by FzK). The sample pebbles were packed in a quartz tube and heated to 900oC under hydrogen atmosphere (1000Pa H2 /Ar) or water vapor atmosphere (50Pa or 200Pa H2O/Ar). Li mass loss was estimated from the change of sample weight before and after the experiment taking account of the release of water vapor that was monitored by hygrometer. The obtained results were compared with that for Li2TiO3 (by JAEA, Li/Ti=2.06, CEA and NFI). It was observed that Li mass loss from samples increased gradually with heating time and tended to approach different constant value for each Li2TiO3 in both hydrogen and water vapor atmosphere. It can be said that each sample has different Li desorption capacity. Li desorption capacity of Li2TiO3 (Li/Ti=2.11) was smaller than that from Li2TiO3 (Li/Ti=2.06). This result indicates that Li addition does not necessarily promote Li mass loss. Li desorption capacity of Li4SiO4 was smaller than that of Li2TiO3. When the amount of Li mass loss is less than the desorption capacity, Li mass loss increased with the increase of the amount of relreased water vapor.

Id 548

Abstract Final Nr. P2.150

Influence of Li mass loss on tritium behavior in Li2TiO3 with excess Li

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Li ceramic breeder materials are placed at high temperature for a long period in DEMO reactor. Therefore, evaluation of Li evaporation amount and its influence on tritium behavior is an important issue. The investigation of Li mass loss in Li2TiO3 with excess Li by Japan Atomic Energy Agency has been proceeded by the present authors. In this study, Li2TiO3 with excess Li, in which Li mass loss of 4 wt% occurred, was exposed to tritiated water and an influence of Li mass loss on tritium behavior was discussed. The Li2TiO3 which was heated at 900oC under 50Pa water vapor for 72hours was used as a sample. BET surface area was measured before tritium experiment. The sample pebbles of 0.5g were packed in a quartz tube and the argon gas containing tritiated water of 705Bq/cc with water vapor of 480ppm was introduced to the sample bed. The tritium concentration in the outlet of the sample bed was monitored by an ionization chamber. The amount of tritium sorbed on the sample was obtained from the change of tritium concentration in the outlet gas. It was found that BET surface area increased from 0.145m2/g to 0.326m2/g with Li mass loss. Tritium sorption amount per surface area [Bq/m2] at 900oC on the Li mass loss Li2TiO3 was same as that on as-received Li2TiO3. This indicates that tritium sorption amount per gram [Bq/g] was doubled as well as increase of surface area. The difference of mass transfer rate in tritium sorption was not observed although there was possibility that microstructure in the pebbles was varied with Li mass loss. It is concluded that tritium sorption capacity [Bq/g] increases with increase of BET surface area [m2/g] but the sorption rate is not changed, even when Li mass loss of 4 wt% occurs.

Id 625

Abstract Final Nr. P2.151

Certification of contact probe measurement of surface wave of Li jet for IFMIF

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The International Fusion Materials Irradiation Facility (IFMIF) is a neutron source for fusion reactor materials. A liquid Lithium (Li) jet with free surface is planned as a target irradiated by two deuteron beam to generate intense neutrons. It is important to obtain information on the surface wave characteristic for safe and efficient neutron generation. Surface wave characteristics experiment using the liquid Li circulation facility has carried out at Osaka University. In our studies, the electro-contact probe apparatus was used for a contact measurement of a surface fluctuation of the Li jet. In this experiment, a liquid Li droplet was observed on the probe, which has a potential of false detection of probe contact signals. False detection might cause errors to results of contact frequency and mean thickness of the Li jet. It is thus important to clarify effect from the droplet formation, in order to ensure measurement accuracy using the probe. In the present study, images are taken by High Speed Video camera synchronized with probe contact signals, and behavior of the Li droplet and signals are evaluated. When a droplet on the probe contacts with the surface, signals which were obviously different without a droplet was observed as a continuous contact for 10 times longer period than contact signals at the position higher than the average thickness. So it is easy to remove false detection at higher position, in fact, the affection to the result of the frequency by probe around average thickness was roughly estimated about 1 %. And the higher the velocity becomes, the more largely they might be affected. These result suggests that surface wave characteristics obtained from probe experiments might be affected by the droplet. Accuracy of contact measurement could be increased by deleting signals which are similar to false signals.

Id 676

Abstract Final Nr. P2.152

Irradiation hardening in pure tungsten before and after recrystallization

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Tungsten (W) is used as armour material for diverter and blanket in fusion reactor. Since the armour receives high heat loading and heavy neutron irradiation, the material may suffer irradiation hardening that is accompanied by loss of ductility. At the same time, a transient flux up to 20MW could cause recrystallization of W. In this research, the effect of recrystallization on the irradiation hardening of W is investigated with focusing on the correlation between microstructure evolution and hardness change. The material used is a pure (99.95wt.%) rolled W. Isochronal annealings for 1 hr and the following Vickers hardness tests were performed to determine the recrystallization temperature of as-received W. Microstructure observations and EBSD analysis were carried out to examine the texture of the as-received W. A larger portion of grain boundaries with the misorientation smaller than 15º existed at the as-received condition but seldom at recrystallized ones. The grain shape aspect ratio is two on the ND-RD and TD-ND plane but isometric on the RD-TD plane at as-received condition. Specimen of two different sampling directions and recrystallized one were irradiated with 6.4MeV Fe3+ to an average 2 dpa at the depth of 600nm beneath the specimen surface at 300, 700 and 1000. Continuous Stiffness Measurement (CSM) of nano-indentation was carried out to measure the irradiation hardening of the irradiated surface layer. Nix-Gao model was used to calculate the bulk equivalent hardness. The results showed that recrystallized W exhibits a higher hardening effect than the as-received ones. The as-received W shows that the irradiation hardening increases as the irradiation temperature increases.

Id 579

Abstract Final Nr. P2.153

Development of iron-base composite materials with high thermal conductivity

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Reduced activation ferritic/martensitic (RAFM) steels, e.g. the Japanese candidate structure material, F82H, developed by Japan Atomic Energy Agency (JAEA), are one of candidate structure materials for Blanket and Diverter in DEMO. Blanket and Diverter will be exposed to the high heat flux from plasma. In particular, Diverter in DEMO could receive more than 10 MW/m2 as well as the diverter in ITER. From this viewpoint, adequate thermal conductivity is requested to structural materials of Diverter. However, the thermal conductivity of RAFM does not meet the demanded performance. In this study, we focused on increasing in thermal conductivity of the complicated joint structure in the heat-resistant equipments. With using a high thermal conductivity material such as CNT or copper, we have developed the iron-based composite materials with a better thermal diffusivity than that of steels. The thermal diffusivity in the iron/CNT composites was not high enough compared with that of pure iron, while iron/copper composite showed a relatively high thermal diffusivity in the joining conditions. One of the reasons not to be improved thermal diffusivity could be non-mono-dispersion of CNT by the formation of carbides in the matrix.

Id 775

Abstract Final Nr. P2.155

Development Status and Strategy of China Liquid PbLi Blanket in China

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Liquid PbLi blanket is one of most promising candidate concept for future fusion power plant. In China, it's being developed by many institutes and universities lead by Institute of Nuclear Energy Safety Technology. R&D activities cover FDS series conceptual designs of LiPb blanket, structural material development, coolant loop technology development, and tritium extraction etc. Conceptual design of Dual Functional Liquid LiPb/He Test Blanket Module (DFLL-TBM) for ITER is being optimized. Also, conceptual designs for China Fusion Experimental Test Reactor (CFETR) and Multi-Functional eXperimental fusion-fission hybrid reactor (MFX) is being developed in INEST. China low activation martensitic (CLAM) steel is selected as the primary candidate structural material for these design and also for Helium Coolant Solid Pebble TBM for ITER of China. R&D activities are being performed for the ITER qualification including several big ingots preparation and supplement of properties database of CLAM. Manufacture technologies of TBM are also being developed including Hot Isostatic Pressing Diffusion Welding (HIP-DW), Electron Beam Welding (EBW) and Tungsten Inner Gas (TIG) welding. And a 1/3 scale DFLL-TBM has been fabricated. A large-scale integrated testing platform including PbLi loop and high pressure helium loop named as DRAGON-V is being constructed, which is aimed at the integrated testing of TBM mockup for PbLi blanket. The functions include magnetohydraudynamic (MHD) effect under high magnetic field and large gap, safety experiments for heat exchanger break, thermal hydraulic for PbLi and helium, and so on. In addition, the test sections could be changed or replaced considered as the different testing purposes and strategies. Progress of these activities and developing strategies for PbLi blanket will be elaborated in the paper.
Keywords: CLAM; TBM mock-up; DRAGON-V

Id 666

Abstract Final Nr. P2.156

Effect of tantalum on the creep rupture properties of clam steel at 823 K

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China Low Activation Matensitic (CLAM) steel is selected as the candidate structural material for Chinese ITER-TBM and FDS serious fusion reactor conceptual designs. Tantalum (Ta) is an essential alloy element for RAFM steels. According to the reduced activation requirement, Ta is used to replace Nb in the F/M steels because it has comparatively lower residual radioactivity. Four 25-kg ingots of CLAM steel named as HEAT 1009A , 1009B, 1009C and 1009D were melted with vacuum induction furnace for different Ta contents i.e. 0.027wt%, 0.078wt%, 0.15wt% and 0.18 wt%, respectively. The microstructure observation indicated Ta-rich carbides restricted the austenite grain growth during normalization and improved the hardness of the steel. Creep tests were carried out at 823 K with the stress of 230 MPa and 240 MPa, respectively. The results showed that the creep rupture life and minimum creep rate increased with Ta content increasing. Key words: CLAM steel; Creep; Tantalum; RAFM steels

Id 467

Abstract Final Nr. P2.157

Microstructure and Mechanical Properties of TIG Welded Joint of CLAM and 316L Steels

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Reduced activation ferritic/martensitic (RAFM) steel is chosen as the primary candidate structural material of fusion blanket for its advantages of void swelling resistance, high thermo-physical and thermo-mechanical properties. As one kind of RAFM steels, China Low Activation Martensitic (CLAM) steel is being developed in the Institute of Nuclear Energy Safety Technology (INEST), Chinese Academy of Sciences (CAS), and chosen as the structural material for FDS series LiPb blankets and China Test Blanket Module (TBM) for International Thermonuclear Experimental Reactor (ITER). According to the ITER design, the TBMs are supported directly by the vacuum vessel, which is made of SS316L(N)-IG (ITER grade). And austenitic stainless steel is to be used as its peripheral equipment material. Therefore, it is necessary to develop dissimilar-metal welding techniques for CLAM and austenitic stainless steels. In this paper, the joint of CLAM and 316L steels was welded by tungsten Inert Gas (TIG) welding with ER309 filler metal, followed by post-weld heat treatment (PWHT) at 1013K for 2h. The microstructure and mechanical properties of the butt welds without obvious defects were investigated before and after PWHT. The hardness distribution in the transverse direction of the joints showed a sharp hardness fluctuation at the fusion boundary region of CLAM steel before and after PWHT, which may be a result of the composition and microstructure discordance in this zone. The tensile strength of the joint was 666MPa at room temperature after PHWT, and the rupture region lied in 316L base metal side. However, the impact energy of 316L heat affected zone decreased by about 100J after PWHT. Furthermore, the toughness of the weld metal was only 106J because of its coarse dendritic austenite. Further experiments to improve the properties of the joints will be carried out to get high quality of the TBM assembly and related components.

Id 435

Abstract Final Nr. P2.158

Fabrication of 1/3 Size CN DFLL-TBM Components with Embedded Cooling Channels

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The Dual Functional Lithium-Lead (DFLL) TBM has been proposed for testing in ITER, and it is under R&D in INEST, CAS (Institute of Nuclear Energy Safety and Technology, Chinese Academy of Sciences) to demonstrate the related technologies of liquid PbLi breeder blanket. In order to effectively transfer the nuclear heat in fusion reactor blanket, the high-pressure helium is selected as the coolant for the cooling channel components of the First Wall (FW), L-shape Cooling Plate (LCP) and Cover Plates (CP) of the TBM. And these components have been designed with thin rectangular cross-section cooling channels, so their fabrication is the key issue of the TBM manufacturing. Two preparation schemes have been adopted for the fabrication process according to the characteristics of these components. Meanwhile, the Electron Beam Welding (EBW) and Hot Isostatic Pressing (HIP) diffusion bonding are applied in the both schemes. In this work, the 1/3 scale prototypes of the components with embedded cooling channels of CN DFLL-TBM have been successfully performed with China Low Activation Martensitic (CLAM) steel. Preliminary non-destructive testing verified the efficiency of the technologies for the component fabrication, and they would be used for the fabrication of the full size components of the DFLL-TBM. Keywords: DFLL-TBM; Cooling channel; CLAM; EBW; HIP

Id 472

Abstract Final Nr. P2.159

Processing and Properties of Tungsten-Steel Composites and FGMs Prepared by Spark Plasma Sintering

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Tungsten is the prime candidate material for the plasma facing components of fusion reactors. For the joining of tungsten armor to the cooling system or support structure, composites or graded interlayers can be used to reduce the stress concentration at the interface. These interlayers can be produced by several technologies. Among these, spark plasma sintering appears advantageous because of its ability to fabricate fully dense parts at lower temperatures than traditional powder metallurgy techniques, thanks to concurrent application of temperature, pressure and electrical current. In this work, spark plasma sintering of tungsten-steel composites and graded layers (FGMs) was investigated. As a first step, pure tungsten and steel powders of different sizes were sintered at a range of temperatures, to find a suitable temperature window for fully dense compacts. Then, composites of several different compositions and from two pairs of powder sizes were produced, as well as FGMs covering the full range from pure tungsten to pure steel. Characterization of the sintered compacts included porosity (by Archimedean method), structure and composition (by SEM and EDS), phase composition (by XRD) and thermal conductivity (by xenon flash method). Furthermore, the capabilities of this technique in joining of bulk materials were explored.

Id 424

Abstract Final Nr. P2.160

Microstructure and mechanical properties of heat-resistant ferritic-martensitic 12% Cr steels

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The mechanisms of structural and phase transformations of heat resistant ferritic-martensitic steels EK-181 (RAFMS RUSFER-EK-181: Fe-12Cr-2W-V-Ta) and ChS-139 (Fe-12Cr-2W-V-Ta-Mo-Nb) have been investigated using transmission electron microscopy, X-ray diffraction, dilatometry and differential scanning calorimetry. The critical points of phase transformations during heating and cooling have been determined for these steels. The characteristic temperature intervals of precipitation of carbides M3C, M23C6, V(CN) have been revealed. It is established that the increase of quenching rate in combination with the stepwise tempering is a promising way to control the parameters of the microstructure and mechanical properties of steels EK-181 and ChS-139. It lead to formation of a microstructure with a high volume fraction of tempered martensite, a high dislocation density in the martensite laths and ferrite grains, and increasing of the volume fraction and dispersion of carbonitride phase V(C, N). After this treatment a significant (up to 820 MPa at T = 20 °C and 420 MPa at T = 650 °C) increase in yield strength of steels under investigation is observed due to the high efficiency both of dispersion and substructural hardening. The features of heterophase and defect substructure of steels EK-181 and ChS-139 after long (13500 hours) annealing at T = 450 °C and 620 °C have been studied. It is found out that a precipitation of stable carbonitride phase V(C, N) has a significant effect on the intensity of martensite tempering developing here. Due to pinning of defects these particles maintain the ferritic-martensitic structure with a high density of defects and relatively high values of short-time strength after annealing at above temperatures. This study has been supported by Russian Foundation for Basic Research under Contract No. 12-03-00488-a.

Id 284

Abstract Final Nr. P2.161

Oxidation behaviour of neutron irradiated Be pebbles

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Beryllium will be used as neutron multiplier material to increase the tritium breeding ratio (TBR) in fusion reactors. Since pure beryllium becomes brittle and swells under neutron irradiation we must understand its physical, chemical and mechanical evolution during irradiation. The chemical reactivity of beryllium is a factor of major concern since the oxidation of beryllium at temperatures above 800 °C can become an uncontrollable process which consumes all the beryllium. As the blanket of the fusion reactor will withstand temperatures in the 600 to 900 °C range it is crucial to understand the chemical behaviour of neutron-irradiated Beryllium under these conditions. In this work we present a detailed study of chemical composition and reactivity of Be pebbles after exposure to neutron irradiation, up to 3000appm He production, during the HIDOBE-01 (HIgh DOse irradiation of BEryllium) campaign in the High Flux Reactor in Petten (HFR). The temperatures during irradiation were in the relevant regime for breeder blanket (425 °C -750°C). The chemical composition of the irradiated samples was studied using ion beam analysis (IBA). Also the possible influence of the irradiation on the oxidation kinetics of Be was assessed. To get information on this issue some samples were oxidized at 700 °C under controlled air atmosphere (60% humidity) and a mixture of 40%N_2 +60%O_2. The thickness of the oxide layer was measured by Rutherford Backscattering Spectrometry (RBS) and the structural changes and surface morphology were followed by SEM. The results show the growth of a protective BeO layer indicating a diffusion mediated process following a parabolic law.

Id 554

Abstract Final Nr. P2.162

TA interdiffusion in W-based composites consolidated by spark plasma sintering

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Tungsten was select for extensive use in nuclear fusion devices due to its high melting point and sputtering threshold as well as low tritium retention [1]. Nevertheless, the application of W in armour components is limited by its high ductile-to-brittle transition temperature (DBTT). A strategy to increase the fracture toughness of W-based materials lies on the development of suitable W-Ta composites following a powder metallurgy route [2]. Tantalum powder/fibre dispersions in a nanostructured W matrix have been produced via spark plasma sintering (SPS) at temperatures ranging from 1500 to 1900 oC. Scanning electron microscopy (SEM), coupled with energy dispersive X-ray spectroscopy (EDS) and electron-backscattered diffraction (EBSD), was used to characterize the microstructure of the composites. The microhardness, thermal diffusivity and geometric density of the consolidated composites were also evaluated. The experimental results revealed that the consolidated materials at lower temperatures (1500 oC) evidenced limited (W,Ta) interdifusion and presented Ta layers surrounding the W nanostructured grains. The composites consolidated at higher temperatures exhibited a binary eutectic between Ta and Ta2O5 phases. The Ta2O5 oxide formed also discontinuous layers at the W/Ta interfaces, allowing for the outer migration of Ta. [1] M. Ubeyli, S. Yalçm, J. Fusion Energy 25 (2006) 197. [2] M. Dias, R. Mateus, N. Catarino, N. Franco, D. Nunes, J.B. Correia, P.A. Carvalho, K. Hanada, C. Sârbu, E. Alves, Journal of Nuclear Materials 442 (2013) 69-74. This work has been carried out in the frame of the Contract of Association between the European Community and Instituto Superior Técnico and was partly supported by the European Communities within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission. Financial support was also received from the Fundação para a Ciência Tecnologia (FCT) grants with references PTDC/CTM/100163/2008 and PEST- OE/CTM-UI0084/2011. The authors acknowledge Fundação para a Ciência e Tecnologia for the grant SFRH/BPD/68663/2010.

Id 123

Abstract Final Nr. P2.163

W-Taf composite materials for nuclear fusion applications

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The high melting point, high sputtering threshold and low tritium inventory rendered W as a potentially suitable material in fusion devices. The major disadvantage of tungsten-grades for plasma facing and structural components in nuclear fusion reactors is the low fracture toughness associated with the high ductile-to-brittle transition temperature. Since tantalum evidences low neutron activation and high radiation resistance, as well as high ductility and toughness relative to W, this metal can be used for W-Ta composites production. Therefore, dispersions of ductile Ta fibres in a W matrix have been proposed as a novel approach for the development of suitable plasma facing materials [1]. In the present study tungsten-tantalum fibres composites (W-Taf) have been produced by pulse plasma sintering (PPS) at 1500oC. The W-Ta composites were implanted with He+ (pre-implantation step) and D+ ion beams at room temperature with fluences in the 1020-1021 at/m2 range. The materials were studied by scanning electron microscopy (SEM) coupled with energy dispersive X-ray spectroscopy (EDS), focused ion beam (FIB), X-ray diffraction (XRD), Rutherford backscattering spectrometry (RBS) and nuclear reaction analysis (NRA). The microstructure observations revealed that after consolidation the W/Ta interface reflects internal oxidation of Ta and no extensive interdiffusion W-Ta was observed. EBSD analysis allowed to identified the presence of Ta2O5 and TaO interlayers witch tend to form between W and Ta. Moreover, blistering occurred in the Ta2O5 and in TaOx regions with single He+ implantation and a more severe effect was observed after the sequential He+ and D+ implantation.

Id 559

Abstract Final Nr. P2.164

Towards the demonstration of industrial production capacity of Spanish ASTURFER ®Ferritic-Martensitic steel at ITER-scale demands

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Reduced Activation Ferritic-Martensitic (RAFM) steels are confirmed as candidate structural materials for ITER Test Blanket Modules (TBMs) and future fusion reactor DEMO. During the last decades diverse R&D Programmes worldwide have established materials specifications for RAFM. The capability to fulfill the EUROFER´s specification (low-activation composition/microstructure/δ-ferrite/precipitates and basic mechanical properties) [ASTURFER®] at laboratory scale was demonstrated by ITMA Materials Technology Institute and Structural Materials Division of Technology Division at CIEMAT in the context of National Programme activities. In the EU, not the expressed but the demonstrated industrial supply capabilities for EUFOFER is today restricted to a limited number of manufacturers for such material under nuclear QA supplying requirements. It could mean Project risks and extra procurements costs. Use of a Vacuum Induction Melting (VIM) scaled furnace is a key QA issue in the material production The present works reports the steps forwards for the certification of the industrial capability (at few Tons scale) of ASTURFER® focusing on: i) the design and engineering of the of the VIM furnace facility and 2) on the key QA production and procurement issues.

Id 1004

Abstract Final Nr. P2.165

Overview of the fusion engineering activities toward DEMO at Rokkasho

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At present, technical R&D tasks mainly on the blanket related materials are on going intensively at the JAEA Rokkasho site under the Broader Approach (BA) framework. Regarding the activities in RAFM, four different heat treatments were tested on F82H-BA12 heat, and tensile and Charpy impact tests are conducted. Some variation is observed, but the tensile strengths are equivalent to those of F82H IEA heats and BA07 heats, and DBTT of F82H-BA12 is better than that of F82H-IEA. On the SiC/SiC composites R&D, test equipment for corrosion of SiC/SiC composites (or SiC) by Li-Pb has been made at ENEA, and will be shipped to Rokkasho in early May 2014. EU-Japan joint experiments will start soon by using it. On the advance neutron multiplier R&D, pebbles of Be-Ti intermetallic compound have been fabricated successfully by with the Rotating Electrode Method using the plasma-sintered Be-Ti rod. Now we are trying to produce Be-V or Be-Zr pebbles which have better nuclear property for tritium breeding ratio. In the tritium technology, tritium durability of Nafion membrane is demonstrated, which is an encouraging result suggesting that Nafion membrane is an adequate material for electrolysis cell of water detritiation system of ITER and DEMO. The samples of JET dust and carbon tiles used in tungsten divertor campaign with deuterium plasma will be shipped to Rokassho soon, and the analyses such as tritium profile measurement will also start. As one of the major fusion engineering activities after BA, a neutron source using an extension of the IFMIF/EVEDA prototype accelerator and the lithium test loop is considered as one of the major activities after the BA activities. Also construction of the test bench for the ITER TBM is planed at Rokkasho. Those plan will be presented.

Id 444

Abstract Final Nr. P2.166

Physical properties of SiC and SiC/SiC composites toward DEMO application

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To assume SiC and SiC/SiC composite materials as functional structures of the fusion DEMO reactor, e.g., flow channel insert application of the DEMO blanket, thermal and electrical insulation is critical requirement and their irradiation tolerance needs to be clarified for the design. In addition, helium/hydrogen peameability is another important fucntion to be required in SiC to consider tritium inventory of the system. To date, there were many researches on irradiation stability on SiC and SiC/SiC composites, specifically focusing on the irradiation-induced microstructure and thermo-mechanical properties, but there is lack of data on physical properties in spite of their importance. Under the Broader Approach activities in Japan, in-situ electrical resistivity and helium/hydrogen permeability measurements are therefore more addressed. The objective of this paper is to evaluate the physical properties of SiC materials including the recent summary of the irradiation experiments to help the DEMO design activities. The radiation-induced conductivity (RIC), which was due basically to the eletrical excitation, was first identified using various irradiation sources such as the gamma-ray and 14 MeV fusion neutron sources and the fundamental data were well-summarized, providing a major conclusion that the RIC depended on the irradiation dose but it was masked by the thermal effects at elevated temperatures. The heavy ion irradiation experiments were also conducted to evaluate the added effects of displacement damage on the electrical conductivity of SiC to assume DEMO relevant environment and the first set of data suggested marked radiation-induced electrical degradation (RIED). In parallel, the diffusion coefficients of hydrogen in SiC were identified, revealing that they were varied depending on experimental methods, i.e., varied diffusion path depending on the microstructure. This paper will finally probide a comprehensive data set for the functional applications of SiC and SiC/SiC composites with remaining issues to be addressed toward the DEMO development.

Id 511

Abstract Final Nr. P2.167

Mechanical properties of TIG and EB weld joints of F82H

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Breeding blanket has box structure and it is necessary to use weld joint for pressure tightness and isotope confinement. Welding of reduced activation ferritic/martensitic steel, F82H for blanket fabrication is intensively investigated under the framework of ITER Broader Approach (ITER-BA). As a R&D activity on materials engineering for DEMO blanket, characterization of F82H weld joints prepared with Tungsten-Inert-Gas (TIG) and electron beam (EB) have been investigated. In this work, 50mm thick plates of F82H were welded using both processes. A similar-metal was employed as a filler for TIG welding. Post-weld-heat-treatment was conducted at 1003K according to the conditions for Grade 91 defined as ASME P-No. 15E, Group No.1. Although the maximum and the minimum hardness of the both joint are similar, the hardness distribution is quite different. The width of EB welds were smaller than that of TIG, and the hardness of EB weld metal was 10% higher than that of TIG. In the TIG welds, the strongest part was heat affected zone (HAZ) heated above phase transformation temperature, Ac1 and the hardness was very similar to the weld metal of EB joint, 280Hv. The hardness of TIG weld metal was around 260Hv. Both welds demonstrated the smallest hardness, 180Hv in the HAZ heated below Ac1 temperature. As a investigation of manufacturing process of box fabrication, second EB weld bead was perpendicularly put on the first EB bead. As a result, the second weld did not weaken the HAZ, but reduced the hardness of the weld metal to 260Hv.

Id 564

Abstract Final Nr. P2.168

Measurement of Li-target thickness in the EVEDA Li Test Loop

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In the International Fusion Materials Irradiation Facility (IFMIF), thickness of the liquid Li plane wall jet must be maintained within 25 ± 1 mm at the nominal velocity of 15 m/s under a vacuum pressure of 10-3 Pa at the nozzle-exit Li temperature of 523 K as a 10-MW deuteron beam target. In the framework of the Engineering Validation and Engineering Design Activities (EVEDA) of IFMIF, we are conducting various experiments in the EVEDA Li Test Loop (ELTL) for the validation of the IFMIF Li loop. The objective of this study is to measure Li-target thickness in the ELTL in the velocity range 10–20 m/s under an argon atmosphere of 0.12 MPa at the operation temperature of 573 K, as a first step to the validation of the stable Li target. The nozzle exit of the ELTL has 100-mm-wide and 25-mm-thick cross-sectional dimension to simulate the IFMIF Li target flow. In the experiment, Li-jet thickness was measured using laser-based distance meter in a area covering the deuteron beam footprint foreseen in IFMIF (approximately 200 mm downstream from the nozzle exit). The measurement precision of the laser distance meter was estimated to be 20 µm. The sampling frequency was set to be 500 kHz. The experimental result showed that time-averaged thicknesses at the beam center was 25.80, 26.06 and 26.09 mm at the velocities of 10, 15 and 20 m/s, respectively. Regarding temporal variation of the thickness, we analyzed time-series data to obtain statistics of wave heights. The results shows that average wave height at 15 m/s was 0.6 mm, which means 0.3 mm in amplitude. Thus, the Li jet under an argon atmoshpere was very stable and satisfied the current design specifiction of the IFMIF Li target.

Id 569

Abstract Final Nr. P2.169

Mechanical properties of the F82H melted in an electric arc furnace

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DEMO reactor requires over 10,000 tons of reduced activation ferritic/martensitic steel (RAFM). Therefore, it is necessary to develop the manufacturing technology for fabricating such large-scale RAFM with appropriate mechanical properties. Through the Broader Approach activity in Japan, we have fabricated some F82H heats applying Electroslag remelting (ESR) process to remove oxide inclusions harmful to the mechanical homogeneity. Through the ESR process, oxide inclusions were successfully removed and appropriate mechanical properties were obtained. In this work, we focused mechanical properties of the F82H-BA12 heat which was melted in a 20 tons electric arc furnace followed by ESR process. Its raw material of iron was blast furnace iron, because electrolytic iron, which has been used in former heats, cannot be used for such large-scale melting due to its limited production volume. After the melting and forging, this F82H-BA12 heat was heat-treated in four different conditions to optimize its normalizing and tempering conditions, and tensile and Charpy impact tests were then performed. The result of these mechanical properties were comparable to those of the former F82H heats less than 5 tons which were melted applying vacuum induction melting.

Id 205

Abstract Final Nr. P2.170

Estimation of the lifetime of resin-insulators against baking temperature for JT-60SA In-vessel coils

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The JT-60SA project is a combined EURATOM(EU) - Japan(JA) satellite tokamak under Broader Approach in support of the ITER project and researches towards DEMO. In-vessel coils such as the Error Field Correction coil and the Resistive Wall Mode control coil are designed and assembled by JA. The resin-insulator is required to have a heat resistance against the baking temperature of vacuum vessel of ~ 200 ºC (40000 hour). Thus the assessment of the heat load is fundamental for the design of the coils. However, the estimation of the lifetime of resin-insulator under the high-temperature region has not been examined. In the present study, the estimation of the lifetime of seven candidate resin-insulators such as epoxy resin, cyanate-ester resin and epoxy-based resin under the ~ 220 ºC temperature region have been performed for the current design of the In-vessel coils. Chemical kinetics of resin has been estimated using Arrhenius equation. Weight reduction of the seven candidate insulators was measured at different heating times under 180ºC, 200ºC and 220ºC environment using three thermostatic ovens, respectively. The reduction of the insulators has been used as input for Weibull-analysis towards Arrhenius-plot. Lifetime of the resins has been estimated for the first time at the high temperature region by the plot. Lifetime of the resin-insulators have been evaluated and discussed as well as the available temperature of the in-vessel coils.

Id 675

Abstract Final Nr. P2.171

Modification of vaccum plasma sprayed tungsten coating on F82H by friction stir processing

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Tungsten (W) is the primary candidate material as a plasma facing material in fusion devices, as for its high melting temperature, good thermal conductivity and low sputtering rate. The vacuume plasma spray (VPS) technique has been investigated as it is practical for coating large area. The issues are the thermal conducitivty and the strength of VPS-W, i.e., the thermal conductivity of VPS-W were significantly lower than that of the bulk W, and the hardness of VPS-W is much less than that of the bulk W. These are mainly caused by the porous structure of VPS-W. In order to solve these issues, friction stir processing (FPS) was applied on VPS-W in this study. The material used as the substrate metal is F82H-IEA heat. VPS coating was counducted to form 0.5 to 2.0 mm thickness of W layers on F82H plates. FPS was performed using a sintered cemented carbide (WC-Co) tool with a flaot bottom face. Various combination of FSP conditions, such as rotating rate, tool travel speed, and load, were examined to obtain the best FSP conditions. The microstrucual observation revealed that a single FSP could disperse almost all pores which was observed in as-VPS-W layer and the grain becomes very fine and isotropic. The double FSP could make remained pores to disappear. Hardness test over FSPed VPS-W layer revealed that the hardness of W becomes equivalent to or higher than that of bulk W. Thermal conductivity of double FSPed VPS-W was examined, and it turned out that its thermal conductivity was about 80% of bulk W at 200°C, and it becomes equivalent to that of bulk W over 800°C. As a whole, it was suggested that FSP can contribute to significant improvement both in mechanical and thermal properties of VPS-W coating.

Id 917

Abstract Final Nr. P2.172

impacts of friction stir processing on irradiation effects in vacuum-plasma-spray coated tungsten and its substrate F82H

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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 steel, with a special emphasis on the impacts of grain-refining induced by FSP. F82H IEA-heat was used as a substrate in this study. Firstly, 2 mm-thick W was coated on F82H by VPS and then modified with FSP. The irradiation specimens were prepared by cutting W-coated F82H into two parts: W and F82H. 18.0 MeV W6+ irradiation was conducted on W to ~5.4 dpa at 800°C at TIARA facility in JAEA, and 6.4 MeV Fe3+ irradiation with/without 1.0 MeV He+ was conducted on F82H to ~20 dpa at 470°C at DuET facility in Kyoto University. Nano-indentation tests were performed to evaluate hardness after the irradiations. Microstructure was characterized by OM, SEM, and TEM. It was revealed that the hardness of the VPS-FSP W after ion-irradiation to 5.4 dpa at 800°C were not remarkably changed, where bulk W usually exhibited significant irradiation hardening.

Id 941

Abstract Final Nr. P2.173

Small Specimen Test Technique and the fusion DEMO strucutral interity evaluation on brittle-ductile transition behavior of RAF/M steels

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The main objective of IFMIF (International Fusion Materials Irradiation facility) is to obtain the material data base obtained from a series of tests using small specimens mainly irradiated in the IFMIF for the design and licensing of fusion DEMO and power reactors. The irradiation volume of the IFMIF is about 13 liters (L) in the total, and the highest displacement damage is more than 20 dpa (displacement per atom)/year in a volume of 0.5 L. Therefore, we have to use small size specimens for it, and small specimen test technique or technology (SSTT) is very important. For mechanical tests using small specimens, high accuracy controllability in the applied stress and displacement is required. In this study, we will present the contents of (1) recent evaluation of SSTT such as fracture toughness and fatigue tests developed in IFMIF/EVEDA project, (2)the evluation of strucural integrity of RAF/M steels on brittle-transition transition behavior for the Fusion DEMO reactors, and (3) engineering desing of the Post Irradiation Examination (PIE) facility of IFMIF. The analysis of the fracture behavior of brittle-ductile transition region of RAF/M steels is very important to evaluate the structural integrity of RAF/M steels for the Fusion DEMO reactors, and it should be applied by new methods such as random-inhomegeneity method of K. Wallin and M. Sokolov and the modified master curve method of ASTM E1921 of P. Spatig, and one of objectives of this study is focued on the further developed analysis to generalize the methods for SSTT and structural integrity of RAF/M steels.

Id 754

Abstract Final Nr. P2.174

Characterization of JET neutron field for material activation and radiation damage studies

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A new Deuterium-Tritium campaign (DTE2) is planned at JET in 2017, with a proposed 14 MeV neutron budget nearly an order of magnitude higher than any previous DT campaigns. With this proposed budget, the achievable neutron fluence on the first wall of JET will be up to about 1020 n/m2, comparable to that occurring in ITER at the end of life in the rear part of the port plug, where several diagnostic components are located. At the expected plasma performance, the neutron flux on the first wall achieves levels comparable to those expected in ITER between the blanket and the vacuum vessel (1017 n/s•m2). This level of neutron flux/fluence will offer the opportunity to irradiate samples of functional materials used in ITER diagnostics, and of materials used in the manufacturing of the main in-vessel ITER components, to assess the degradation of the physical properties and the neutron induced activities, respectively. The purpose of the present work is to characterize the neutron and gamma ray field inside the JET device during DT plasma operations. An analysis of the neutron/gamma ray flux, energy spectrum and dose rate levels is performed at selected irradiation locations, such as the neutron activation irradiation ends, the new long term irradiation stations located inside the vessel and inside a circular horizontal port, where samples would be exposed to the maximum neutron flux or fluence. Other locations are considered around the vacuum vessel and inside a vertical port where active tests on fiber optics and on dielectric materials would be performed. The neutron flux and spectrum at different irradiation ends are calculated and compared. The study is performed for selected materials at these locations, for which other important quantities – nuclear heating and the radiation damage – are calculated with the use of the MCNP code and the appropriate nuclear data libraries.

Id 913

Abstract Final Nr. P2.175

Effects of Sc addition on the mechanical properties of RAFM steel

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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 the blanket is limited by the high-temperature creep and low-temperature irradiation embrittlement of the structural material. Accordingly, it is essential to develop RAFM steel that is able to withstand high temperatures and high-energy neutron irradiation. Since 2012 Korea has developed its own RAFM steel for test blanket module (TBM) application in international thermonuclear experimental reactor (ITER). The present work examine the effects of Sc on mechanical properties of RAFM steel. 9Cr-based RAFM steels with Sc contents varying 0 to 0.05 wt.% were designed and produced by vacuum induction melting, and then formed into a plate by hot rolling, which is followed by various austenitization and tempering treatments. A series of mechanical tests was carried out to determine the tensile, impact, and creep properties of tempered plate. It is found that addition of Sc up to 0.03 wt.% increases the yield strength but further addition reduces strength. The creep resistance is also enhanced by addition of Sc and its dependence on Sc contents is similar to that found for the yield strength. Impact resistance is, on the other hand, slightly impaired by Sc addition. As the solubility of Sc in Fe is quite high (about 0.6wt.%), it is thought that addition of Sc increases the strength of RAFM steel by solid solution hardening, which in turn affects the creep and impact resistance.

Id 376

Abstract Final Nr. P2.176

TSTA Piping and Flame Arrestor Operating Experience Data

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The Tritium Systems Test Assembly (TSTA) was a facility dedicated to tritium handling technology and experiment research at the Los Alamos National Laboratory. The facility operated from 1984 to 2001, running a prototype fusion fuel processing loop with ~100 grams of tritium as well as small experiments. There have been several operating experience reports written on this facility's operation and maintenance experience. This paper describes analysis of two additional components from TSTA, small diameter gas piping that handled small amounts of tritium in a nitrogen carrier gas, and the flame arrestor used in this piping system. The operating experiences and the component failure rates for these components are discussed in this paper. Comparison data from other applications are also presented. This work was prepared for the U. S. Department of Energy, Office of Fusion Energy Sciences, under the DOE Idaho Operations Office contract number DE-AC07-05ID14517.

Id 639

Abstract Final Nr. P2.177

Modeling an unmitigated thermal quench event in a large field magnet and its busbar

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The superconducting magnet systems of future fusion reactors, such as a Demonstration Power Plant (DEMO), will produce magnetic field energies in the 10s of GJ range. The release of this energy during a fault condition can produce arcs that can damage the magnets of these systems. A recent example of such events occurred in CERN's Large Hadron Collider (LHC) [1]. The public safety consequences of these events must be explored for a DEMO reactor because the magnets are located near the DEMO's primary radioactive confinement barrier, the reactor's vacuum vessel (VV). Great care will be taken in the design of DEMO's magnet systems to detect and provide a rapid field energy dump to avoid any accidents like the one that occurred at LHC. However, given recent events in the nuclear power industry, even beyond extremely unlikely events (frequency 1x10-6/yr) must be considered for DEMO safety assessments. One such accident is an unmitigated quench event. During this event's progression, the fault condition proceeds undetected, with the potential of producing melting of the magnet. If molten material from the magnet impinges on the walls of the VV, these walls could fail, resulting in a pathway for release of radioactive material from the VV. A model is under development at Idaho National Laboratory (INL) called MAGARC to investigate the consequences of this accident in a large toroidal field (TF) coil. Recent improvements to this model include the solution of Maxwell's Equations to predict the magnitude of eddy currents produced in the magnet as the magnetic field decays. This model is described in this paper along with predictions for a DEMO relevant unmitigated quench event in a toroidal field magnet. [1] L. Rossi, Superconductor Science and Technology, 23 (2010).

Id 852

Abstract Final Nr. P2.178

Development of Virtual Reality-Based Simulation System for Nuclear and Radiation Safety and Its Application

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Due to the risks involved, work scenario in nuclear and radiation environment is always designed based on experts' and past experience, without considering human error by unskillful handing, risks associated with unpredictable situation. So the suggested work scenarios are always not the optimal scenario according to as low as reasonably achievable (ALARA) principle. Virtual reality (VR) technology, in turn, has been applied in many diverse areas, with the possibility of performing virtual simulations of real environments. Then, it is possible to simulate these risk situations considering hypothetical scenarios. Based on VR technology and high-precision whole-body computational phantom, a virtual reality-based simulation system for nuclear and radiation safety named RVIS. The basic functions of RVIS include: (1) CAD-based physics modeling and virtual assembly simulation of complex components; (2) virtual roaming simulation and organic dose evaluation; (3) visualized analysis of dynamical 3D radiation field coupled with geometry model. In comparison with other similar systems, the improvement features of RVIS are that it allows the accurate assessment of organic dose rate based on accurate voxel human phantom, and advanced real-time visual analysis of multiple physics calculation result. RVIS makes it possible to safely perform work scenario designs, optimization and training of workers in risky areas. In this paper, the system architecture, ALARA evaluation strategy and some advanced visualization methods used in RVIS are described. At the same time, virtual reality-based simulation for ITER PF repair has been shown for verification of the system, including GUI-based maintenance scenario design, advanced visualized analysis of calculation result, real-time organic dose assessment, virtual roaming simulation and maintenance scenarios optimization and analysis.

Id 966

Abstract Final Nr. P2.179

neutronics modeling of blankets for iter c-lite based on mcam

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ITER is a complex, multi-national fusion project. In order to meet the subtleties and strict requirements of its nuclear analysis, detailed neutronics reference models of the Tokamak machine need to be established. To establish the latest model ITER C-lite, detailed structure of components such as blankets, are required to be described and updated. However, the blankets consist of high-order spline surfaces as well as complex structures, which bring great difficulty for neutronics modeling. MCAM developed by FDS Team, is designed to be a Multi-Physics Coupling Analysis Modeling System. It is an advanced modeling program aiming to solve modeling problems for multi-physics simulation. It is capable to convert models between CAD systems and multiple Monte Carlo codes including SuperMC, MCNP, PHITS, TRIPOLI, FLUKA and Geant4. It can significantly reduce the manpower and enhance reliability for constructing simulation code input files with complex geometry and it has been used by many institutions in more than 40 countries. MCAM has been widely applied in ITER modeling. And the neutronics models of blankets were created for ITER C-lite based on MCAM. In these models, spline surfaces were replaced by analytic surfaces, unnecessary detail structures were removed, and materials were defined based on the latest design. Validation testing showed the correctness of the generated blankets neutronics models. These models will be incorporated into ITER C-lite model for nuclear analysis.
Keywords: MCAM, CAD, nuclear analysis, modeling

Id 969

Abstract Final Nr. P2.180

3D Visual post-processing for nuclear analysis based on RVIS

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Nuclear analysis is becoming more and more data-intensive with increasing complexity of problems and requirement on high accuracy. Huge simulation result data makes data post processing and analysis more difficult and inefficient, so it is necessary to find efficient and effective approach for huge data post processing and analysis. RVIS, developed by FDS Team, China, is a virtual-reality based simulation system for nuclear and radiation safety. Based on advanced data visualization and computer graphics technologies, RVIS provides following visual nuclear analysis capabilities: 1) directly support of data post processing for multiple codes, such SuperMC, MCNP and TORT. 2) Multiple data visualization including contour, 2D section map, 3D iso-surface, and 3D mesh map. 3) Data visualization coupled with calculation geometries, such as 2D map coupled with geometry with frame, and mapping result data onto geometry surface. 4) Visualization of operation process and real-time dose assessment. Many visual nuclear analysis cases have been performed based on RVIS. And in this paper, 3D visual nuclear analysis of occupational radiation dose during ITER PF4 maintenance was presented. This case shown that RVIS provides a useful and effective approve the efficiency of nuclear analysis.

Id 961

Abstract Final Nr. P2.181

First experimental results of particle re-suspension in a low pressure wind tunnel applied to the issue of dust in fusion reactors

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For safety reasons, it is essential to quantify the resuspended dust fraction that can be potentially mobilized in ITER during a loss of vacuum accident (LOVA) with air or water vapour (coolant event) ingress. Following our sampling achieved in ASDEX Upgrade tokamak (see companion paper, Rondeau et al., SOFT 2014), we got information on the particle characteristics representative of those anticipated in ITER, and potentially mobilizable by an airflow. Here, to validate a resuspension model taking into account the effect of low pressure, we provide new experimental data quantifying the fraction of particles mobilized by airflow. The experiments were carried out in the Aarhus Wind Tunnel Simulator AWTS-II. It has an internal volume of 35 m3 and is housed at Aarhus University, Denmark. The dust deposition was performed by sedimentation at low pressure (< 60 mbar) in a smaller wind tunnel simulator. The dust used was aluminium oxide powder: one with a mass median diameter of 9.9 µm and one of 17 µm. In addition, we used a tungsten powder with a mass median diameter of 10.6 µm. The dust deposition densities were between 0.1 and 1.3 mg.cm-2 for the alumina powders and between 2.5 and 40 mg.cm-2 for the tungsten powder. The experiments were performed at 1000, 300, 130 and 10 mbar. After setting an environmental pressure in the wind tunnel, the airflow velocity was increased by steps at 10 minutes interval. At the step-end, the resuspension fraction was measured by an optical technique based on measuring the transmitted light intensity through the dust deposit. To show the influence of pressure, we present here an example of three pressure / velocity values (respectively) which are expected for 10% resuspension of alumina powder (9.9 µm): 300 mbar / 18 m.s-1; 130 mbar / 29 m.s-1; 10 mbar / 59 m.s-1. As expected, the mobilization of particles at low pressures requires a higher airflow velocity but, as the Knudsen number increases, the expression of lift and drag forces (α pv^2) must be corrected accordingly (transition between viscous and molecular flow).

Id 839

Abstract Final Nr. P2.182

Characterization of tungsten particles in AUG tokamak which are potentially mobilizable by airflow

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During the normal operating condition of the future ITER tokamak, a massive production of dust in the toroidal vacuum vessel is anticipated. This dust, originating from the erosion of the tokamak internal walls by the plasma, would be mobilized to some extent during a loss of vacuum accident (LOVA). For safety reasons, it is essential to quantify the resuspended dust fraction that can be potentially mobilized during such an event. In these studies, we provide new data about the dust characteristics, pertaining to the mobilization mechanism and specific to each fusion reactor. We collected dust particles deposited in the vacuum vessel of the German tokamak AUG (ASDEX Upgrade). Albeit much smaller than ITER, AUG design includes a characteristics particularly relevant to our study: its plasma facing components are made of tungsten, which will also be the component making up the ITER divertor. Moreover, AUG toroidal chamber, at a smaller scale (45 m3 vs. 1400 m3), has the same magnetic configuration than ITER. The particles were collected by means of a device (the so-called 'Duster Box', Gensdarmes et al., SOFT 2012) which allows the resuspension of particles from a surface by a controlled air-flow with calibrated friction velocity. The particles thus collected were characterized by means of an optical microscope (Morphologi G3 Malvern) combined with an acquisition system which allows picture analysis. Thus, we gained information on the particle size distribution but also on the morphology of the particles. The microscope detected a total of 70,391 particles, coming from an auscultated surface of 126 cm2, showing a bimodal distribution with a first mode composed of flakes at 0.7 µm and a second mode composed of spherical particles at 1.9 µm. This study on ITER-like particles, which are potentially mobilizable by an airflow in the tokamak, paves the way for a better analysis of the consequences of a LOVA (see companion paper, Rondeau et al., SOFT 2014).

Id 839

Abstract Final Nr. P2.183

Stellarator-Specific Developments for the Systems Code PROCESS

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In order to study and design next-step fusion devices such as DEMO, comprehensive systems codes are commonly employed. The code package PROCESS is such a tool, widely used for Tokamak systems studies. In this work, the application of PROCESS to Stellarators is addressed. In order to incorporate a Stellarator module in the system code PROCESS, Stellarator-specific models are proposed which reflect the differences due to the confinement concept. The identified and subsequently developed models include: - a geometry model based on Fourier coefficients which can represent the complex 3-D plasma shape, - a basic island divertor model which assumes diffusive cross-field transport and high radiation at the X-point, - a coil model which combines scaling aspects based on the Helias 5-B design in combination with analytic inductance and field calculations, and - a transport model which employs a predictive confinement time scaling derived from 1-D neoclassical and 3-D turbulence simulations PROCESS provides a framework for finding a self-consistent operating point satisfying physics and engineering constraints. It also provides common routines for non-device-specific systems such as plant power balance, and routines for optimisation of the plant based on figures of merit. This approach is investigated to ultimately allow one to conduct Stellarator system studies and to facilitate a direct comparison between Tokamak and Stellarator DEMO and power plant designs in a common framework. Current results of such a comparison are presented which demonstrate the viability of the Stellarator concept for the same set of requirements.

Id 124

Abstract Final Nr. P2.184

Water and air ingress accident transients in fusion facility ITER: Source term analysis

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Amongst the accidental scenarios that can lead to environmental contamination in fusion facilities, losses of containment due to an air ingress and leak of coolant systems in the vacuum vessel (VV) have been identified as reference ones. IRSN started a few years ago a research program devoted to the development of from one side thermochemical data and from the other side accident simulation code to evaluate the consequences of such design basis situations. This code named ASTEC also has the capabilities to evaluate the dynamic loading into the VV and the Vacuum Vessel Pressure Suppression System (VVPSS) in these conditions. Significant steps have been recently reached in this strategy that will be presented here. The most important contributors to the environmental and health consequences would be co-deposited tritium, tokamak activated dust (Be and W), tritiated water and activated corrosion products from the cooling circuits. Important lack of knowledge was identified in the thermochemical properties of some of beryllium oxides and hydroxides that have been reduced using an innovative theoretical approach. On the other side, IRSN has extended the modelling capabilities of ASTEC to account for all the system included in the second containment barrier. ASTEC is ready to simulate accident scenarios and consequences using simple correlation models for the filtering systems. The transport and chemical speciation inside first and second confinement barriers will be presented. The impact of the dust initial state on the distribution and speciation at the end of these accidental transients will be also shown. Next step will be once they will have been determined to develop predictive models for all these filtering systems. The capability easily to integrate models for one given system and to analyze subsequent effect on all reference accident progress and consequences can be particularly interesting in the phase of the ITER realization where the design is not definitively fixed. It could also be powerful for the design of the foreseen DEMO facilities.

Id 50

Abstract Final Nr. P2.185

Chemical reaction of lithium with atmosphere containing variable humidity at room temperature

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In the International Fusion Materials Irradiation Facility (IFMIF), a back plate of the target assembly will be exchanged during the in-service period. During the works, the lithium components will react chemically with the surrounding atmosphere. In this research, the chemical reaction of lithium in air, oxygen and nitrogen containing variable humidity at room temperature has been investigated to estimate the chemical reaction during the exchange works. In air and oxygen environments, the weight gain increased linearly with the elapsed time increased. In the same elapsed time, the weight gain increased with the humidity increased. No difference was observed between in air and in oxygen. In nitrogen environment except the 75% of relative humidity (RH), the weight gain was accelerated with the elapsed time, and the acceleration decreased with the RH increase. The rate of weight gain at 25%-RH was the highest in all nitrogen tests. The rate decreased with the humidity increased, and the rate at 75%-RH was the same as those of air and oxygen environments at 75%-RH. It was considered that the increase of the weight gain observed in air- and oxygen-tests was caused by the lithium hydroxide formation. In nitrogen environment, although the weight gain rate increased by the moisture in low humidity condition, it decreased when more moisture was supplied. It seemed that the behavior of the weight gain increase observed in low humidity region was caused by the acceleration due to lithium nitride formation. When more moisture was supplied, the main reaction with lithium would be change from lithium nitride formation to lithium hydroxide formation. It was considered that since main chemical reaction was lithium hydroxide formation in high humidity environment in all cases, the weight gain rates of all gas environments were close in all RH75% tests.

Id 371

Abstract Final Nr. P2.186

Design study for DEMO concept definition

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For ensuring operation reliability and safety, we have focused on the concept development of a low power DEMO with a fusion power (Pf) of 1.35-1.5 GW, major radius (Rp) of about 8.2 m and average neutron wall load of less than 1 MW/m2. The blanket concept is based on pebble bed breeder cooled by pressurized water of 290 – 325ºC and 15.5 MPa. For the capability of plasma current ramp-up and short pulsed operation by central solenoid (CS), Rp is considered to be about 8.2 m to provide a sufficient CS installation space. Divertor simulation and safety analysis suggest that Pf of 1.35-1.5 GW can provide a feasible solution regarding divertor heat removal and safety. A divertor simulation confirmed that the divertor peak heat load could be less than 6 MW/m2 for the conventional divertor configuration at Pf of less than 2 GW when fully detached plasma is maintained. In the case of Pf = 1.35 GW, an analysis for a postulated total loss of coolant accident (LOCA) indicated that the blanket surface temperature rise due to residual heat would be is lower (1,350ºC) than the metling temperature of a Reduced Activation Ferritic/Martensitic (RAFM) steel even if no active measures of enhancing heat removal of are taken. Based on safety analysis, a design philosophy of the confinement of tritium and other radioactive materials in the DEMO plant was determined. The management value of tritium concentration in the primary cooling water is an important trade-off issue closely related with safety and the net electricity production. A design approach for defining the management value of concentration is presented in terms of public dose assessment for a severe ex-vessel LOCA and the electric power assessment required for a water detritiation system in the primary water.

Id 672

Abstract Final Nr. P2.187

Analyses of iron and concrete shielding experiments at JAEA/TIARA with JENDL/HE-2007, ENDF/B-VII.1 and FENDL-3.0

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IAEA released a new Fusion Evaluated Nuclear Data Library, FENDL-3.0, in December of 2012. FENDL-3.0 extends the neutron energy range from 20 MeV to 150 MeV. Now there is increasing interest in nuclear data above 20 MeV. Thus we have analyzed the iron and concrete shielding experiments with the 40 and 65 MeV neutron sources at TIARA in Japan Atomic Energy Agency with the latest high-energy nuclear data libraries, JENDL/HE-2007, ENDF/B-VII.1 and FENDL-3.0. The Monte Carlo code MCNP-5 and ACE file of JENDL/HE-2007, ENDF/B-VII.1 and FENDL-3.0, which are supplied from JAEA, BNL and IAEA, respectively, were used for this analysis. The collimated neutron beam and experimental assemblies were modeled in the analysis. The measured source neutron data were adopted in the analysis. The followings are found out from the results. 1) The calculations with JENDL/HE-2007 agree with all the measured ones well. 2) Those with ENDF/B-VII.1 tend to overestimate the measured ones with the thickness of the assemblies largely. 3) Those with FENDL-3.0 agree with the measured ones well for the iron experiment, while they overestimate the measured ones well for the concrete experiment largely. Some data in ENDF/B-VII.1 and FENDL-3.0 should be revised.

Id 850

Abstract Final Nr. P2.188

Dynamic Modelling of Balance of Plant Systems for aPulsed DEMO Power Plant

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The current baseline concept for a European DEMO defines a pulsed reactor producing power for periods of 2-4 hours at a time, interrupted by dwell periods of approximately half an hour. Such a power generation scheme will be unlike that of any other thermal power station. The heat transfer system and power generation equipment will be required to cope with load changes that are not only frequent, but also significant in magnitude and rapid in time, and will need to demonstrate sufficient resistance to cyclic fatigue to provide safe and reliable performance. Thermal energy storage systems have been proposed to mitigate some pulsing issues;however, this will simultaneously increase the cost and complexity of the balance of plant. The impact of time-variant effects on the balance of plant must therefore be assessed. This work presents the development of a dynamic model of the primary heat transfer system and associated steam plant for a water-cooled DEMO, capable of simulating plant operation over a number of pulses of power output. The model has been created in the Modelica programming language, using the ThermoPower library of thermodynamic components. During the dwell period, the temperature at localised points within the steam generator is found to change by at most 15°C, while pressure transients in the primary coolant are kept below 0.35 MPa. This is achieved by defining system behaviour such that the primary coolant flows continuously throughout the dwell period while the secondary steam flow is reduced, minimising the heat extracted from the primary circuit and maintaining the average coolant temperature. Thermally induced stresses on the steam turbine rotor are estimated using a simplified model, and found to be small. If the turbine is kept spinning to also minimise mechanical cycling, pulsed operation of a water-cooled DEMO without thermal energy storage may be feasible. This work was funded by the RCUK Energy Programme under grant EP/I501045 and the European Communities under the contract of Association between EURATOM and CCFE. This work was carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Id 1110

Abstract Final Nr. P2.189

The Monte Carlo approach to the economics of a DEMO-like power plant.

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The optimized design of the European demonstration power plant, DEMO, is being derived from several technical and physical guidelines but also the economic aspects will affect the project definition. An early assessment of the impact of specific design choices on both the construction cost and of the cost of electricity of a DEMO-like commercial power plant would lead to project arrangements that would ensure a higher economic competitiveness of fusion. For this purpose an innovative approach is used here to overcome the uncertainties on both the design of DEMO and the costs of the components. Specifically, a DEMO-like power plant is modeled with FRESCO, a System Code developed by Consorzio RFX, and the assessment of the most critical technical, physical and economic parameters is carried out through the Monte Carlo method. The effects on the cost of electricity of the technical life of replaceable components, such as blanket and divertor, of the time required for their replacement, of the technical life of the power plant itself, of the lead time, of the discount rate are investigated. The construction costs, defined as direct costs, are instead studied as a function of the cost of materials and/or the power plant components. The outcomes are density probability curves that indicate the range of possible value of the cost of electricity and the corresponding probability. It turns out that the financial aspects have the largest impact on the cost of electricity. Similarly the uncertainties about the future cost of materials due to the recently experienced cost escalation have a significant effect on the construction cost. The power plant is supposed to operate in steady state mode but insights on the pulsed mode are also given. This work outlines a methodology to be used concurrently with the DEMO design activities to optimize the economics of a future commercial power plant.

Id 524

Abstract Final Nr. P3.001

Comparison of Particle Image Velocimetry Flow Data inside HyperVapotrons with Computational Fluid Dynamics

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HyperVapotrons are two-phase water-cooled heat exchangers able to receive high heat fluxes by employing a cyclic phenomenon called the “Vapotron Effect”. Although initially designed by Thomson CSF for cooling Klystron tubes, HyperVapotrons have been used routinely in high heat flux nuclear fusion applications, for example as neutral beam stopping elements. A detailed experimental investigation on the cyclic “Vapotron effect” giving rise to the ability to sustain in steady state heat fluxes in excess of 10MW/m2 has not yet been possible and hence the phenomenon is not yet well understood. The coolant flow structures that promote the effect have been a major point of interest, and many investigations based on Computational Fluid Dynamic (CFD) simulations have been made in the past. The understanding of the physics of the coolant flow features inside the device may hold the key to further control of the complex mechanisms responsible for improved heat transfer and allow optimisation of engineering designs. However, past computational investigations have not been experimentally evaluated. The work described in this paper aims to do so by employing cold flow Particle Image Velocimetry (PIV) measurements of the fluid flow structures in HyperVapotron optical models with high spatial resolution (30µm). The same measurements are subsequently calculated under the same boundary conditions via commercial CFD software. The CFD calculation is then compared to the experimental velocity measurements to be able to deduce the accuracy of the CFD investigations carried out. The investigation is aiming to evaluate the accuracy of the CFD simulations of the fluid flow for several flow conditions and geometries and quantify the effect of the computational grid resolution on the computed flow field. This unique comparison between computational and experimental results in HyperVapotrons will direct future efforts in analysing similar high heat flux devices by either computational or experimental methods.This work was funded by the RCUK Energy Programme and EURATOM

Id 1033

Abstract Final Nr. P3.002

Design and Testing of Electromagnetic Pump in PREKY-I Facility

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Force driving pump is a key sub-system which makes the circulation reality for liquid LiPb(lithium-lead) experimental loops . And the three-phase cylindrical linear inductor electromagnetic pump(EMP) is considered as the promising choose for the loop operation because of its advantages of no leakage and simple structure etc.. In China, related to the engineering of LiPb blanket for ITER , the Dual-Functional Lead-Lithium Testing Blanket Module (DFLL-TBM) was designed and optimized for many years by FDS team in INEST (Institute of Nuclear Energy Safety Technology), Chinese Academy of Sciences. In the paper, the EMP was designed with the theory and numerical simulation method .The structural parameters(polar distance, pump ditch, magnetic core clearance and pitch etc.) of the EMP were calculated using the method of theoretical calculation according to the required parameters (flow, pressure, temperature, conductivity, viscosity and density etc.). Meanwhile, the designed EMP performances were tested basing on ANSOFT MAXWELL software .The simulation results showed that the EMP could meet the requirements of liquid LiPb experimental loop DRAGON-V, in which integrate test would be carried out as well as the engineering of the DFLL-TBM concept investigated. The hydraulic and mechanical performances of EMP were tested in the liquid metal test facility named PREKY-I. The test results showed that the EMP worked at good state when the working current was up to 170 ampere, consequently, the flow rate reaced 5m^3/h, and pressure head 7.5bar when the outlet temperature was kept to 380^o during the test, which was close to the expected design parameters. And the EMP had run for 200hrs continuously in order to ensure the stable performance. With the test results, the EMP could be determined as the DRAGON-V loop drive, whose lifetime and the stable performanes test would be done and tested further in future. Keywords: EMP; Design and testing; DFLL-TBM; PREKY-I

Id 399

Abstract Final Nr. P3.003

Installation and Commissioning of the Negative Ion Optimization Experiment

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Negative ion sources are key components of the neutral beam injectors for thermonuclear fusion experiments. The NIO1 experiment is a radio frequency ion source generating a 60kV-135mA hydrogen negative ion beam. The beam is composed by 9 beamlets over an area of about 40 x 40 mm2. This experiment is jointly developed by Consorzio RFX and INFN-LNL, with the purpose of providing a test ion source, capable of working in continuous mode and in conditions similar the ones foreseen for the larger ion sources of the Neutral Beam Injectors for ITER. At present research and development activities address several important issues related to beam extraction, optics, and optimization; the modular design of the NIO1 experiment allows for quick replacement and upgrading of parts. This contribution presents the installation phases and the status of the experiment, the commissioning of the plant units, such as cooling system and power supplies, and of some diagnostic devices, like the beam dump and spectroscopy. Examples of the first commissioning pulses will also be shown.

Id 697

Abstract Final Nr. P3.004

Development of 3D ferromagnetic model of tokamak core with strong toroidal asymmetry

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Effect of presence of ferromagnetic material in assembly of magnetic fusion devices is conventionally investigated as an influence of tokamak iron core on plasma equilibrium, under assumption of full toroidal symmetry. However, such an approach is insufficient to characterize influence of tokamak core or of ferromagnetic inserts on toroidal asymmetries of tokamak magnetic fields (e.g. resonant magnetic perturbations or toroidal field ripple), which are in the center of current interest. In previous work [1], toroidally symmetric iron core model was introduced for tokamak GOLEM, together with constraints of its applicability. Proposed contribution presents newly-developed 3D model of arbitrary ferromagnetic elements, based on the same physical principles – screening of currents along ferromagnetic boundary [2]. Model is benchmarked using series of new measurements of magnetic fields affected by influence of strongly-asymmetrical tokamak core. Due to presence of magnetic flux leaks from tokamak core, effect of stray fields on measurements and implementation of gaps into model are discussed as well. Moreover, principal results from previous work [1] are re-examined from the viewpoint of 3D geometry. [1] T. Markovic, M. Gryaznevich, I. Duran, V. Svoboda, G. Vondrácek, Evaluation of applicability of 2D iron core model for two-limb configuration of GOLEM tokamak, Fusion Engineering and Design 88 (2013). [2] M. Gryaznevich, T.G. Kilovataya, V.N. Pyatov, Effect of ferromagnet on the equilibrium of a tokamak plasma, Soviet Journal of Plasma Physics 9 (1983).

Id 777

Abstract Final Nr. P3.005

Status and perspectives of the ASDEX Upgrade gas inlet system

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Present-day fusion devices using metal plasma facing components require a versatile gas inlet system for operation. Supplying the plasma discharge with hydrogen isotopes, providing nitrogen or noble gases for radiation cooling and intended charging of impurity gases are tasks of such sophisticated systems. For dedicated investigations or avoiding mutual interference with diagnostics widely distributed positions for the gas valves are advantageous. To match these partly conflicting requirements the gas inlet system (GES) of ASDEX Upgrade has been substantially renewed and upgraded. The plasma discharge gas feeding is done by fast piezo valves using an integrated pressure sensor and a closed loop controller for flow measurement. This enables calibrated gas flows on a time scale of some milliseconds. To reduce electromagnetic disturbance a custom-built connecting cable between controller and valve was installed combining both actuating and signal voltage wires. The refurbishing of the entire GES started by replacing parts of the PLC system with decentralised periphery inside the torus hall including rewiring and renewal of tubing for the auxiliary pneumatic valves. Galvanic isolation is achieved by connecting these components via optical fibre. Many of the presently 19 fast piezo valves were relocated and equipped with a tube on the vacuum-side for pinpoint gas delivery. Now eight valves feeding the lower divertor, four the low field side mid-plane and also four the upper divertor are available. Three valves are installed for special applications as for example the injection of expensive tracer gases. Each of the valves is supplied independently by a valve matrix where one out of 10 different types of gases can be selected. A change of gas species is usually possible between two consecutive plasma discharges. This paper will give a comprehensive overview of the project, the operational limits and advantages of the new system.

Id 272

Abstract Final Nr. P3.006

Continuity and Enhancement of Quality Management during commissioning of W7-X

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Wendelstein 7-X, the first numerically optimized stellarator, is presently under construction at the Max-Planck-Institut for Plasma Physics (IPP) in Greifswald. The assembly of the essential components like the central machine and the heating systems as well as the diagnostics and the peripheral systems will soon be finished. Therefore now the commissioning is prepared as a new phase in the project. For the general planning and execution of the commissioning, the entire Wendelstein 7-X system is divided into appropriate components, subsystems and systems. The approach for taking this system into operation will lead to so called local commissioning which is usually executed for individual systems in connection with the necessary peripheral devices and auxiliary systems. The local commissioning will be carried out in parallel to the last assembly activities. The subsequent step-wise testing and commissioning of the systems in connection with the central device of W7-X including the central safety control and the central data acquisition system is performed in the second step, the so called integrated commissioning phase. It leads directly to the preparation of first plasma operation. Also the Quality management system, which is certified according to ISO 9001, has to be adapted to these new conditions. On the one hand the continuity of the running system must be maintained; on the other hand new aspects must be taken into account. Following this intention some new elements are added, and existing, proven tools are modified. The paper will describe the major organizational structure and tasks and the quality planning and assurance tools in more detail and will summarize the experience during the first commissioning steps of the vacuum and cryogenic systems.

Id 155

Abstract Final Nr. P3.007

Features and Analyses of W7-X Cryostat System FE Model

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The Wendelstein 7-X stellarator is presently under construction 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), ports, thermal insulation, vessel supports and the machine base (MB). The main task of the cryostat system is to provide an insulating vacuum for the cryogenic magnet system while allowing external access to the PV through ports for diagnostic, supply and heating systems. The cryostat is subjected to different types of loads during assembly, maintenance and operation. This ranges from basic weight loads from all installed components to mechanical, vacuum and thermal loads. To predict the behaviour of the cryostat in terms of deformations, stresses and support load distribution a finite element (FE) global model has been created called the Global Model of the Cryostat System (GMCS). A complete refurbishment of the GMCS has been done in the last 2 years to prepare the model for future applications. This involved a complete mesh update of the model, an improvement of many model features, an update of the applied operational loads and boundary conditions as well as the creation of automatic post processing procedures. Currently the GMCS is used to support several significant assembly steps of W7-X that involve the cryostat system, e.g. the removal of temporary supports beneath the MB and the transfer of the PV from temporary to the final supports. In the next months the model will also support the commissioning of W7-X which includes the first evacuation of the OV and PV.

Id 236

Abstract Final Nr. P3.008

Status of the ITER Vacuum Vessel Construction

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The ITER Vacuum Vessel (VV) has major functions of being the first confinement barrier and removing nuclear heating during plasma operation. The VV has been designed as a fully welded torus-shaped, double wall structure with in-wall shielding (IWS), and cooling water between the shells in order to satisfy the technical requirements. In accordance with French regulation the VV and ports are classified as Nuclear Pressure Equipment due to the presence of radioactive products in the plasma chamber and wáter-cooled structure. The VV structurally supports all in-vessel components, such as blanket modules, divertor cassettes, port plugs, in-vessel coils and diagnostics, etc. Five Procurement Arrangements (PAs) with ITER Domestic Agencies (DAs) have been signed for the fabrication of nine sectors (seven sectors by the EU DA and two sectors by the KO DA), IWS (IN DA), upper ports (RF DA), and equatorial & lower ports (KO DA). The VV is under manufacturing by four Domestic Agencies after completion of engineering designs that have been reviewed and agreed by the Agreed Notified Body. Manufacturing designs of the VV for regular sectors for the EU and KO DAs have been completed, component by component, by accommodating requirements of the RCC-MR 2007 edition. Manufacturing of the VV first sector has been started in February 2012 in Korea and for the In-Wall Shielding in May 2013 in India. EU starts manufacturing of its first sector from February 2014 and Russia starts the Upper port manufacturing from May of 2014. All DAs have manufactured several mock-ups including real-size ones to justify/qualify and establish manufacturing techniques and procedures. They have performed required qualification and justification of manufacturing through review and approval process by the IO and ANB. All required documents for start of manufacturing including manufacturing procedures and manufacturing inspection plans, etc, have been developed and approved prior to start of manufacturing. In this paper, the status of the ITER VV design finalization and fabrication will be described.

Id 153

Abstract Final Nr. P3.009

Optimisation of CAD methodology for the final design of complex vacuum system components

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There are a number of large complex mechanical components within the ITER Vacuum System, for which “build-to-print” designs are required to be delivered for “In Kind” procurements. The design for these components has to be fully specified as the performance responsibility lies with ITER. The approach is that the provided “build-to-print” design are used to develop fabrication and workshop drawings, suppliers can however request design changes which have to be approved at ITER. Such components include the Torus, Cryostat, and Neutral Beam Cryo-pumps, these components perform functions which are key for ITER’s mission and also important for safety and hence the design configuration needs to be effectively managed throughout the product life cycle. The tools used for CAE and CAD include CATIA, for design development, ENOVIA, for lifecycle management and ANSYS for structural analysis. How the design, models, drawings and analysis are structured and organised using these tools is key to being able to efficiently manage the design through the lifecycle. To develop an effective approach various design standards and guidelines were reviewed. In addition the experience of design development in manufacturing from issuing a “build to print” design through to “as-built-drawings” is considered with different approaches of different manufacturing companies. In this paper as an example an overview of the CAD methodology used for the development of the Neutral Beam Cryo-pump is presented. The methodology applied gave the flexibility required by the maturity of the fabrication method, possibility to iterate on the design without complete redesign, changing interfaces and elimination of possible errors which can appear when different designers work on the same components. It is shown how splitting the CATIA products by functional role (e.g. supporting structures, pressure equipment, vacuum flanges) allows iterating during the development and manufacturing, to integrate modifications generated by structural analysis or manufacturer’s feedback.

Id 184

Abstract Final Nr. P3.010

Overview of the ITER Tokamak Complex Building & Integration of Plant Systems towards Construction

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The ITER Tokamak Complex consists of Diagnostic Building, Tokamak, & Tritium buildings. The Tokamak Basic Machine is located in the bioshield pit of the main nuclear building. Plant systems are implemented in the 3 buildings and are strongly interfacing with the Tokamak. The reference baseline configuration is a set of height thousand CAD models that defines in an exhaustive way the layout of tokamak and plant systems The last 4 years, one of the main ITER challenges was to improve the maturity of the plant systems layout in order to confirm their integration in the building final design and freeze the interface definition in-between the systems and to the buildings. The propagation of safety requirements in the design of the nuclear building like confinement, fire zoning and radiation shielding was of first priority. A major effort was placed by ITER Organization together with the European Domestic Agency (F4E) and the Architect/Engineer as a joint team to fix the interfaces and the loading conditions to buildings. A set of 80 thousands anchor plates was implemented in the detailed design of the three concrete buildings covering seven the levels and roof. Embedded Plates are positioned with respect to concrete reinforcement definition and are sized on the basis of gravity and seismic loads issued by plant systems. The most demanding systems in terms of interface definition are Water Cooling, Cryogenic, Detritiation, Vacuum, cable trays and building services. Systems Penetrations and temporary openings for installation are all defined. Some mandatory Project Change Requests (PCR) were implemented in a tight allocated time schedule: The most demanding one was to implement a much more robust design of the Tokamak cryostat supporting system. The 18 supporting columns (2001 baseline) were replaced in 2012 by a concrete crown linked to basemat and bioshield together with a set of 18 sliding bearings. That major change was implemented successfully in the building construction design to allow basemat construction phase starting in 2013. The paper gives an overview of the nuclear buildings final configuration and highlights the large progress made on the final integration of systems in the Tokamak Complex. The revised design of the Tokamak machine supporting system is described too.

Id 735

Abstract Final Nr. P3.011

The Design of HFTM of IFMIF

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Under Broader Approach (BA) Agreement between EURATOM and Japan, IFMIF/EVEDA (International Fusion Materials Irradiation Facility/Engineering Validation and Engineering Design Activities) has been performed since the middle of 2007. The 'Intermediate IFMIF Engineering Design Report' was completed in June 2013. This paper is summarizing the engineering design of the high flux test modules (HFTM) and the PIE facilities within the IFMIF/EVEDA project. In HFTM, Vertical Type and Horizontal Typehave been designed. The HFTM is exchanged regularly every approximately one year, and it is loaded again by the new HFTM of the new specimen or the reloaded specimen and can continue an irradiation examination again. It is obtained the specimen which simulated the irradiation damage of the level up to the design life year of the nuclear fusion reactor by this repetition. In HFTM, it is designed to simulate the content of materials He and H as an irradiation examination condition of materials for nuclear fusion reactors other than irradiation temperature and irradiation damage rate dpa. A sample specimen for various materials examinations is loaded in HFTM. SSTT (Small Specimen Test Technique) is applied to the specimen. It is shown the kind of a loaded specimen and an example of the number. Various materials evaluation examinations are carried out in the post irradiation examination room. It is shown an example of testing equipment for materials examinations and the examination evaluation apparatus for PIE (Post Irradiation Examination). The irradiation examination test plan is discussed as an example of the irradiation examination test plan. The validation activities are on-going and potential lessons learnt could be implemented upon the accomplishment of the validation activities. After these irradiation examinations plan in IFMIF, it is expected that the necessary irradiation materials characteristic data sets for DEMO reactor would be maintained systematically.

Id 500

Abstract Final Nr. P3.012

In-Pile Lithium-Lead Loop

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Russian Federation as a participant of ITER project takes part (on the rights of Partner) in the development of Lithium-Lead Ceramic Breeder Test Blanket Module (LLCB TBM). Lithium metatitanate (Li2TiO3) is used as breeder material and Lithium Lead eutectic (Li17Pb83) is used simultaneously as coolant, neutron multiplier and moderator. The critical issues connected with Li17Pb83 during testing in ITER are as follows: 1) tritium generation (breeding) and extraction, 2) compatibility of eutectic alloy with structural material; 3) bred tritium permeability through the structural material. In order to resolve experimentally the above mentioned issues the project materials on the in-pile loop with Li17Pb83 (on the IVV-2M fission reactor base) have been developed. In order to validate the in-pile liquid metal loop the out-of-pile facility with liquid Li17Pb83 and the section for Li17Pb83 alloy preparing have been fabricated. The brief descriptions of IVV-2M fission reactor and in-pile loop with Lithium Lead eutectic are presented in this article. Also this material carries the information on the out-of-pile facility with Li17Pb83 and on the section for Li17Pb83 alloy preparing.

Id 343

Abstract Final Nr. P3.013

Overview on ITER and DEMO relevant blanket fabrication activities of the KIT INR and related frameworks

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Fabrication experiments have been carried out in the KIT with the goal to qualify manufacturing technologies for the realization of Fusion reactor components. The main focus of the activities in the Institute of Neutron Physics and Reactor Technologies (INR) has been on the Test Blanket Module (TBM) for ITER. Sets of Fabrication and Welding Procedure Specifications have been demonstrated and qualified in relevant scale for TBM structural and functional components. The activities have been organized in two different frameworks: i) A national program for development and qualification of industrial fabrication technologies funded by the German governmental institution BMBF (Bundesministerium für Bildung und Forschung) related to fabrication developments for all relevant sub components (First Wall, Breeder Zone Cooling Plate and Stiffening Plate) and assembly technologies (Laser, EB, Tig- welding in all TBM relevant plate thickness combinations) ii) A procurement contract with F4E with the goal to provide a full-scale feasibility mock up for a HCPB TBM Breeder Zone cooling plate incl. the related fabrication specification. Both projects are scheduled to be concluded in 2015 where the fabrication related activities will be continued with focus on DEMO relevant manufacturing studies and feasibility experiments in the framework of the Breeder Blanket Project in Eurofusion Consortium. This paper summarizes achievements in fabrication of ITER and DEMO relevant blanket sub components and assembly technologies of the running out projects. Blanket related technologies in relevant scale which have been developed in the present frameworks are referenced. Also the contribution identifies interfaces and interactions in between the different development approaches of the past, present and future.

Id 882

Abstract Final Nr. P3.014

IFMIF-test facilities: functional analysis and improvement of hot cells

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The International Fusion Material Irradiation Facility (IFMIF) is dedicated to study and qualify structural and functional materials for use in future fusion power plants. During the Engineering Validation and Engineering Design Activities (EVEDA) phase, preparation for construction is the top task. Test Modules of different types are housing numerous small specimens. Inside the Test Cell they are irradiated with high energy fast neutrons (up to 40MeV) for 11months and receive a damage rate up to 20dpa. After irradiation, maintenance takes place for one month. Nearly all of the test modules are replaced. The irradiated modules are transferred to hot cells, where they are disassembled to gain around 1000 mm-size specimen. They get cleaned and most of them are sent out for further investigation whilst some of them are prepared for an additional irradiation phase. Disassembling work includes cutting the Test Modules into big pieces to get the specimen containing part. This work requires low precision tools but produces big amounts of waste, contamination, and etc. After that, fine scale work on the specimen containing parts takes place, with high precision tools, small amounts of waste and very few contamination. Specimen planned for recycling have to be re-inserted into specimen-containers and new test modules have to be assembled, both under hot-cell-conditions but in "clean" environment. Being the result of a carefully performed functional analysis of the sequence of process steps along the flow of specimen (logistics process), it was applied to the former existing reference design which had foreseen only one big hot cell to perform all the processes. As consequence, the hot cell area was divided into different hot cells according to the different functions described above. Alongside, also the optimized arrangement of the cells to provide fast and high efficient work on the modules is described in this paper.

Id 825

Abstract Final Nr. P3.015

Study of wettability of Eurofer steel by the liquid metal sodium-potassium (NaK-78)

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The main mission of the International Fusion Materials Irradiation Facility (IFMIF) is to provide irradiation conditions, similar to those of the DEMO fusion reactor, for several testing capsules packed with specimens of Eurofer steel which is the main candidate as a structural material for the first wall and blanket of the fusion reactor. Using the liquid metal sodium-potassium alloy (NaK-78) to fill the small gaps among the Eurofer specimens was introduced to improve the thermal conduction among the specimens and achieve uniform temperature distribution. Hence, the wettability of Eurofer specimens by NaK is investigated by two experiments to evaluate the success of this concept. In the first experiment, a stack of IFMIF Eurofer specimens is packed in a steel capsule that has two sides of glass to observe the NaK filling process by video recording. The specimens and NaK are heated up to 400ºC by two heaters attached to the capsule. The NaK flow in the capsule and wettability of the specimens are evaluated. The second experiment has two Eurofer strips, with a gap between them, placed in a temperature-controlled crucible filled with NaK to study effect of: (i) NaK temperature, (ii) surface roughness of the Eurofer strips, and (iii) gap size on the wettability of these Eurofer strips. The results of these experiments are summarized and discussed in this paper.

Id 958

Abstract Final Nr. P3.016

Experimental activities at the TUD-NG 14 MeV neutron generator facility in support of the European fusion technology program

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The neutron laboratory of Technical University of Dresden has been involved in the European fusion technology program for many years. Experimental activities comprise mock-up and material activation experiments to provide a database for checking nuclear data used in radiation transport and dose rate calculations. Recently focus has shifted to the development and testing of neutronics instrumentation for the European test blanket modules (TBM) for ITER. We present a technical description of the neutron generator of the laboratory, its accelerator, tritium target assembly and monitoring system. The neutron source is a solid-type water-cooled tritium target based on a titanium matrix on a copper carrier. The neutron yield at a typical deuteron beam current of 1 mA is of the order of 1011 n/s in 4π. A pneumatic sample transport system is available for TBM neutronics instrumentation development and nuclear cross section measurements. A standard tool for the analysis of these experiments is the MCNP code. Previously a tabulated neutron source descriptions in the MCNP input file was used, however, this approach is not flexible if beam parameters etc. change. We have therefore investigated and compared the application of a MCNP extension for the description of the DT neutron source developed at FNG of ENEA Frascati as well as the application of the MCUNED extension to MCNPX. For the latter, deuterium cross section data for the d-d and d-t reactions from the ENDF/B-7.1 library have been processed into an ACE file at KIT. Calculational results have been checked with experimental data from dsimetry foil irradiations at different angles with respect to the deuterium beam of the neutron generator. Reasonable agreement has been found, however some improvement is still underway.

Id 1011

Abstract Final Nr. P3.017

Conceptual Design Study of the K-DEMO Magnet System

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As the ITER is being constructed, there is a growing anticipation for an earlier realization of fusion energy. A major design philosophy for the initiated conceptual design study for a steady-state Korean fusion demonstration reactor (K-DEMO) is engineering feasibility. A two-staged development plan is envisaged. K-DEMO is designed not only to demonstrate a net electricity generation and a self-sustained tritium cycle, but also to be used, in its initial stage, as a component test facility. Then, in its second stage, a major upgrade is carried out by replacing in-vessel components in order to show a net electricity generation on the order of 500 MWe. After a thorough 0-D system analysis, the major radius and minor radius are chosen to be 6.8 m and 2.1 m, respectively. In order to minimize wave deflection, a top-launch high frequency (200 GHz) electron cyclotron current drive (ECCD) system will be the key system for the current profile control and off-axis current drive of K-DEMO. For matching the high frequency ECCD, a high toroidal field (TF) is required and can be achieved by using high current density Nb3Sn superconducting conductor. The peak magnetic field reaches to 16 T with the magnetic field at the plasma center above 7 T. As advantages of using high magnetic field, the operation limits of maximum plasma current and density can be increased and also a higher fusion power can be achieved with a same reactor construction cost. The K-DEMO magnet system consists of 16 TF coils, 8 central solenoid (CS) coils, and 12 poloidal field (PF) coils. Internally cooled cable-in-conduit conductors (CICC's) are used in all of the K-DEMO coils. Key features of the K-DEMO magnet system include the use of two TF coil winding packs, each of a different conductor design, to reduce the construction cost and save the space for the magnet structure material. The conceptual design of K-DEMO magnet system is presented together with preliminary design parameters.

Id 224

Abstract Final Nr. P3.018

System analysis of the requirements to the IGNITOR tokamak site location

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At present time there is an active discussion between Italian and Russian Parties to expedite the idea to jointly realize the IGNITOR project. As a location for the IGNITOR tokamak the Russian Party suggested the TRINITI site, which is situated near Moscow (now it is the territory of “big Moscow”). One of the most important tasks among initial tasks to realize the project is to determine the technical requirements for power supply and engineering infrastructures to provide correct exploitation of IGNITOR tokamak and researches realization according to the scientific program. For these purposes the system analysis of the technical requirements to the IGNITOR tokamak site location was carried out. The in-depth system analysis of the description of the Ignitor project developed by Prof. Bruno Coppi and his team over a long time was applied during this work. The dates for this analysis were extracted from big numbers of the papers, published in some scientific journals and presented at international conferences. The number of the supporting engineering systems and subsystems was determined. The technical requirements for these systems and subsystems were established too. The received technical requirements will be used at nearest future to provide the development of the preliminary concept project of the modernization and reconstruction of power supply and engineering systems of the TRINITI site under IGNITOR project requirements.

Id 842

Abstract Final Nr. P3.019

Final Design And Test Results Of A High Voltage Amplifier For A Gyrotron Body Power Supply

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The design of a high voltage amplifier is being developed that uses series-connected transistors to control the output [1]. Although having many potential applications, this amplifier is designed to meet at a minimum the requirements of a body power supply for depressed collector gyrotrons that are used in electron cyclotron systems on the DIII-D tokamak and other devices worldwide. Therefore, the output voltage needs to be adjustable up to 35 kVDC and be capable of square-wave modulation at frequencies up to 5 kHz. Continuous operating body currents for a gyrotron are on the order of a few tens of milliamps. However, to attain the required speed for modulation with load capacitances as high as 2.5 nF, the ability to both source and sink peak currents as high as 5 to 10 A is required. The design of a 40 kV, 250 mA high voltage amplifier has now been completed and the amplifier was fabricated. The transistors are configured into modules and the modules are mounted in four card cages within an enclosure that has a size equivalent to a standard electronics rack. The high voltage DC power supply, that provides the input voltage to the amplifier seciton, and the controls for the amplifier are housed within a separate electronics rack. An 800 kΩ, 2.5 nF dummy load for testing the amplifier was also fabricated, and the amplifier was been tested into this load up to its full 40 kV output. The description of the recently completed amplifier and the test results will be presented. [1] J. Tooker, P. Huynh., J. Fusion Eng. Des., 88 (2013), 521

Id 118

Abstract Final Nr. P3.020

Modeling and experimental studies of the DIII-D neutral beam system

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The Neutral Beam system on DIII-D consists of eight ion sources on four beamlines. The basis of the system is the Common Long Pulse Source (CLPS). Essentially 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. There are a number of proposals to build on the recent very successful DIII-D off-axis neutral beam (OANB) upgrade that involved tilting a beamline to provide off-axis current drive. One future upgrade proposal includes increasing the injected power and energy by extending the beam pulse length and increasing the beam voltage. Another proposal is to reconfigure one beamline to give it the capability of co- and counter-injection as well as off-axis injection. In this paper we present the results of beam physics experimental and modeling efforts aimed at learning from and building on the experience of the DIII-D OANB upgrade and other NB system upgrades such as those at JET. The modeling effort includes electrostatic accelerator modeling (using a Poisson solver), gas dynamics modeling for the neutralizer and beam transport models for the beamline. Experimentally, spectroscopic and calorimetric techniques are used to evaluate the system performance. We seek to understand and ameliorate problems such as anomalous power deposition, originating from misdirected or excessively divergent beam particles, on a number of beamline components. We qualitatively and quantitatively evaluate possible project risks such as neutralization efficiency deficit and high voltage hold off associated with increasing the beam energy up to 105 keV. Work was supported by the US Department of Energy under DE-DC02-04ER54698.

Id 659

Abstract Final Nr. P3.021

Research of inverter type high voltage power supply with duty cycle modulation for neutral beam injector

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To construct a high power negative neutral beam injector (NBI) system, an ultra-high voltage power supply for acceleration grid is required. The inverter type high voltage power supply, based on inverter, step-up transformer and diode full bridge rectifier, is the advised scheme for ITER NBI system. The acceleration grid power supply (AGPS) of ITER NBI is include five stages, each of which is an inverter type power supply with an output voltage of 200kV. This paper investigates the inverter type power supply based on neutral point clamped (NPC) three phase three level (TPTL) inverter and three phase step-up transformer, focusing on the voltage regulation and ripple with duty cycle modulation of the inverter, considering the transformer connection and leakage inductance. Because of the high voltage of DC-link, the NPC TPTL inverter is proposed for the AGPS. The voltage regulation with different transformer connections, wye-wye and delta-wye, is compared initially. The transformer is considered to be ideal with no leakage inductance at first. Analysis indicates that the voltage ripple can reduce to half, when replacing wye-wye connection with delta-wye connection, if the duty cycle of inverter is larger than 1/3. The transformer turn ratio can be decreased too, which may decrease the leakage inductance. Therefore the delta-wye connection is advised. Then the influence of the leakage inductance of transformer is discussed. This inductance may decrease the output voltage, however, on the other hand, it can mitigate the voltage ripple. The circuit mode is described, presenting the relationship among the duty cycle, inductance and output voltage. Simulation result is given, to verify the theoretical analysis above. The leakage inductance is proposed to be 8%, in case of 150Hz square wave modulation and 0.47uF output capacitor, to optimize the voltage loss and ripple.

Id 336

Abstract Final Nr. P3.022

A PXI controller for PSM high-voltage power supply on J-TEXT

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A 100 kV/60 A high-voltage power supply (HVPS) has been designed and built on J-TEXT tokamak to energize the gyrotrons of the electron cyclotron resonance heating (ECRH) system. The HVPS is based on pulse step modulation (PSM) technology, composing of 144 identical switch power supply (SPS) modules in series. For the sake of guarantee the normal operation of HVPS and satisfying the requirement of gyrotrons, a controller based on PCI eXtensions for Instrumentation (PXI) technology has been developed to operate the insulated-gate bipolar transistors (IGBTs) which are used as switches in SPS modules and realize the control algorithm. The PXI controller can communicate with the J-TEXT Control Data Access and Communication (CODAC) system as a client-side of the Experimental Physics and Industrial Control System (EPICS), which conforms to the ITER CODAC standard. The PXI controller communicates with all the modules via 288 optical fibers to control the output voltage, and monitor the working state of each module. The PXI controller has precise synchronization, the delay between output signals is less than 10ns. The response time for overcurrent fault is about 500ns. The experimental results shows that the HVPS performs well under the control of the PXI controller. The output voltage of the HVPS reached 100 kV/30 A with ripples <1% and overcurrent response time of 5μs. With an improved control algorithm, the PXI controller can rebalance the output voltage and frequency in 60 μs when individual module malfunctions during operation. Thus, the propoosed controller based on PXI technology can fulfill the isolation, control, and protection requirements of the HVPS.

Id 498

Abstract Final Nr. P3.023

Development of a high power wideband polarizer for electron cyclotron current drive system in JT-60SA

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A wideband polarizer which consists of a twister and a circular polarizer has been developed for an electron cyclotron current driving system in JT-60SA, where the output frequencies of a dual frequency gyrotron for JT-60SA are 110 GHz and 138 GHz. As a first step, the grooved mirrors in a miter bend polarizer were optimized for dual frequencies with numerical simulations using a FDTD method. As a next step, the polarization properties of a twister and a circular polarizer were measured at the dual frequencies in low power tests. After that, the groove depth of the circular polarizer was modified for considering the difference between the numerical simulation and the experimental result. Poincaré spheres were almost covered with the combined operations of the twister and the circular polarizer at the dual frequencies. The thermal stress and maximum temperature of a polarizer at an input power of 1 MW, a pulse duration of 100 s were estimated by numerical simulations using a finite element method. Finally, the twister polarizer has been tested at an input power of 0.25 MW during 3 seconds at a frequency of 110 GHz. The rf power and the pulse length were restricted by the conditioning period. The dependence of the twister ohmic loss on an input polarization and a mirror rotation angle agreed with the theoretical prediction, qualitatively.

Id 904

Abstract Final Nr. P3.024

Progress in Negative Ion Source, ROBIN, operation at IPR

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The Indian program on Negative Ion Beams (NNB) R&D has begun with the single driver based Radio Frequency (RF) ion source, set up under a Licentiate arrangement with IPP Garching. This experimental system, ROBIN, has delivered negative ion beams in the volume mode for current densities of >5 mA/cm2, with an electron current to ion current ratio of ~ 2. Optimisation of the operational parameter space has been carried out for operation of this ion source at pressures ~ 0.3 Pa and input powers up to 80 kW. A limited extraction area has been applied for this optimization, to ensure compatibility with the limited current rating of 400 mA available in the 10 kV power supply for extraction. The mandate is to reach 30 mA/cm2 in the Cesiated source and the availability of the full rating of 35 A/ 11 kV Extraction Power Supply System (EPSS) has enabled the operation of ROBIN for the realization of its mandated parameters of a beam current of 6 A, for an accelerating voltage of up to 40 kV, with the integration of the 35 kV/15 A Accelerator Power Supplies (APSS). A pre-requisite for this operation is the operation of the ion source with Cesium delivered at a rate of ~ 10 mg/hr from an indigenously built Cesium oven. The operation is supported by a fully integrated ~350 channel Data Acquisition and control system and probe, spectroscopy (including laser photo-detachment) and a 40 channel current interceptor calorimeter diagnostics for the measurement of plasma and ion beam parameters, for the surface conversion mode of operation. This campaign concludes the realization of ROBIN mandate, the results of which would be the subject of this paper.

Id 947

Abstract Final Nr. P3.025

Technological and Physics Assessments on Heating and Current Drive Systems for DEMO

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(8) Centre de Recherches en Physique des Plasmas, EPFL, Lausanne, Switzerland
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The physics requirements on DEMO heating and current (H&CD) systems are often beyond the actual level of design maturity and technology readiness required. The recent EU fusion roadmap advocates a pragmatic approach and favours, for the initial design integration studies, systems to be as much as possible, extrapolated from the ITER experience, and on the use of materials adequate for the expected level of neutron fluence. To meet the goal to demonstrate the production of electricity in a DEMOnstration Fusion reactor (DEMO) with a closed fuel cycle by 2050, the right balance must be found between high reliability, availability, maintainability, inspectability (RAMI) on the one hand and good performance, efficiency and optimized design on the other hand. In the work-programme 2013 of the Power Plant Physics & Technology (PPP&T) a number of studies were performed. The outcome of these investigations is presented in this paper. The starting point of work was performed with physics simulations, from 0D-codes like PROCESS, which include the whole DEMO fusion power plant site, and 1D-codes like ASTRA/TRANSP to derive the 1D kinetic plasma profiles. The four H&CD systems Neutral Beam Injection (NB), Electron Cyclotron (EC), Ion Cyclotron (IC) and Lower Hybrid (LH) were considered for further investigations, mainly on the question of optimization of efficient current drive. In parallel the (i) technological maturity was considered (e.g. 240 GHz gyrotrons for EC) and wherever necessary (ii) improvements (e.g. multi-stage depressed collector for EC) and (iii) novel solutions (e.g. photo-neutralization for NB or new antennae concepts for IC) were proposed to overcome the limitations of the present H&CD systems with respect to DEMO plant requirements. Further constraints imposed by remote maintenance (RM) or breeding blanket (BB) interactions are described.

Id 849

Abstract Final Nr. P3.026

Progress in the ITER Electron Cyclotron Heating & Current Drive System Design

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An Electron Cyclotron (EC) system is one of four auxiliary plasma heating systems to be installed in ITER tokamak. The ITER EC system consists of 24 gyrotrons (RFPS; RF power source) with associated 12 high voltage power supplies (HVPSs), a set of evacuated transmission lines (TLs) and two types of launchers. The whole system is designed compatible with propagation of 170 GHz of up to 20MW microwave power into the plasma. The primary functions of the system include plasma start-up, central heating and current drive (H&CD) and magneto-hydrodynamic (MHD) instabilities control. The design takes present day technology and extends toward high power CW operation, which represents a large step forward as compared to the present state of the art. The ITER EC system will be a stepping stone to future EC systems for DEMO and beyond. The EC system is faced with significant challenges, which not only includes an advanced microwave system for plasma heating and current drive applications but also has to comply with stringent requirements associated with nuclear safety as ITER became the first fusion device licensed as basic nuclear installations as of 9 November 2012. Since conceptual design of the EC system established in 2007, the EC system has progressed to a preliminary design stage in 2012, and is now moving forward toward a final design. The majority of the subsystems are getting to knuckle down the detailed design to realize the future advancement envisioned toward the final design completion.

Id 351

Abstract Final Nr. P3.027

Status of R&D Activity for ITER ICRF Power Source System

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India is in-charge for the procurement of ITER Ion Cyclotron Resonance Frequency (ICRF) sources (1 Prototype + 8 series units) along with auxiliary power supplies & Local Control Unit. Based on overall ITER ICRF source requirement (2.5 MW per source at 35-65 MHz/CW/VSWR 2.0), which is very stringent in nature, specifications are generated for various sub-systems/components etc. As there is no unique amplifier chain able to meet the output power specifications, the layout consists of two parallel three-stage amplifier chains, with a combiner circuit on the output side. This kind of RF source will be unique in terms of its stringent specifications and building a first of its kind is always a challenge. Constraints on ICRF source components are identified, in particular concerning the final stage tube of the amplifier. An R&D phase has been initiated for establishing the technology related to this package, considering single amplifier chain experimentation (1.5MW/35-65 MHz/3600s/VSWR 2.0). In 2012, ITER-India has signed two contracts, one with Continental Electronics Corporation (US) and another one with Thales Electron Devices (France) for establishing the technology in very high power RF amplifiers, using different type of vacuum tubes (Tetrode & Diacrode). The contracts are to design & develop driver and final stage amplifiers. Tubes and cavities will be integrated in full amplifier chain and tests under ITER specifications will validate each design. To test the performance of the amplifier chains, high power Test Rig (3MW/CW capability) is being developed at ITER-India lab. The same test rig will generate matched as well as mismatched situation as required for ITER scenarios. This paper will describe the present status of R&D activity to resolve technological challenges involved in operation of very high power RF sources and various infrastructures developed at ITER-India Lab to support such operation.

Id 346

Abstract Final Nr. P3.028

Design of Vacuum Vessel for Indian Test Facility (INTF) for 100KV Neutral Beams

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INTF Vacuum Vessel is designed to install a full-scale test set-up of Diagnostic Neutral Beam (DNB) for the qualification of beam parameters and the behaviour of beam-line components prior to installation and operation in ITER. Vacuum vessel is designed in cylindrical shape having length of ~9m with diameter of ~4.5m and has a detachable top-lid for mounting as well as removal of internal components during installation and maintenance phases. The Vessel has hemispherical dish-ends with large openings for High-Voltage Bushing on one side & Duct on another side. Vessel main shell is provided with openings for hydraulic, cryo, gas-feed and diagnostics. Vessel duct is composed of three segments with length ranges from 3m to 5m with diameter of ~1.5m and one vessel at the end to house the second calorimeter. The objective of this paper is to present the design and analysis of Vacuum Vessel, with respect to functional requirement. The design calculation for the thickness, support and nozzle-flange & their reinforcement are done as per ASME-BPVC SectionVIII-Div.1 and subsequently Finite Element Analysis method has been adopted to verify the design. The main challenge for design was to limit the deflection within 1mm near beam-source support structure. The design is optimised to keep the main vessel structure safe with the thickness of 15mm with 7 stiffeners. Ducts & 2nd calorimeter vessel have the thickness of 10mm with 2 stiffeners. The hemispherical dish-ends have thickness of 12mm. The stress limit & functional requirement of having <5mm deflection over the vessel & <1mm deflection at the Beam Source support locations are verified by ANSYS results. After design validation, the INTF vessel is in manufacturing phase now, so this paper also presents its manufacturing plan, fabrication challenges & the current status of vessel fabrication.

Id 952

Abstract Final Nr. P3.029

Design modification of ITER equatorial EC launcher for electron cyclotron heating anc current drive optimization

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An ITER equatorial EC launcher has been design to inject a 170GHz, 20MW millimeter (mm) wave beam to plasma. A toloidal steering of the beams ($20º \leq Q_T \leq 40º$) was also required and the launcher design successfully could perform with the required function with the beam steering by the combination of three movable mirrors. The recent physics analysis [1] reveals that there is the potential to more than double the drive current at the range of r = 0.4~0.6 if a beam steering direction can be modified from toroidal to poloidal. However, the change of steering direction has a significant impact on the design of the Equatorial EC Launcher (EL), especially, mm-wave propagation and blanket shield modules (BSMs) and a knock-on effect impacting the internal shield structure The EL has three sets of mm-wave beam row and each is composed of eight waveguide lines and a quasi-optical transmission region that is formed a steering and fixed mirror, locating in front of the waveguide outlet. It has been successfully performed that the outstanding configuration of the mirrors is developed, the mirror actuator is changed from the push-pull to pneumatic concept and the modification of the internal shield structure is minimized as much as possible. This creative modification ensures that both mm-wave beams from the middle and bottom row pass through the same BSM opening and then, the feasible and reliable design of BSMs can be carried out. The nuclear analysis of this design modification shows that the residual dose rate at the launcher back end is reduced by 20%. This paper summarizes that the possible solution of the mm-wave design modification enhancing the off axis current drive functionality is developed and ensures the effective mm-wave propagation, feasible design of the EL BSMs and nuclear shield structure.

Id 527

Abstract Final Nr. P3.030

Experiments of High-Power Multi-Frequency Gyrotron and Long Dsitance Transmission

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ITER electron cyclotron system is designed to inject RF beam of 20 MW by using 24 sets of 170 GHz / 1MW gyrotrons. In JAEA, developments of high-power and long-pulse gyrotrons are underway. At present, we are developing a gyrotron with high-order mode (TE31,11) to reduce heat load in a cavity resonator and achieve the output power of more than 1MW. This gyrotron has advantages for not only the heat load, but also multi-frequency oscillation. RF beams of 170 GHz, 137 GHz (TE25,9) and 104 GHz (TE19,7) are also oscillated by varying a magnetic field strength at the cavity resonator. Measured radiation profiles and the positions at the front of a diamond window agreed with the calculation results. In high power experiments (2 s), the output power, the oscillation efficiency and the electric efficiency for 170 GHz, 137 GHz and 104 GHz oscillation are 1.2 MW / 27 % / 43 %, 870 kW / 26 % / 41 % and 750 kW / 25 % / 37 %, respectively. At present, the pulse length achieved 224 s with 800 kW and 440 s with 650 kW, 1000 s with 380 kW for 170 GHz oscillation. In order to suppress RF loss in equatorial / upper port launchers, a HE11 mode purity of more than 95 % is required at MOU exit. By adjusting two mirrors in MOU, the HE11 mode purity achieved 94.5 % at the MOU exit. An experiment of ITER relevant transmission line (TL) was carried. The TL distance is 40 m and it includes six miter bends, two waveguide switches, an expansion waveguide, an exhaust waveguide and two gate valves. The axis of waveguide components was aligned to coincide with a light axis of a laser. As the result, the mode purity of more than 91 % was attained before an equatorial launcher mock-up. Then, the transmission experiment in the mock-up is planned.

Id 473

Abstract Final Nr. P3.031

Mechanical and quasi-optical design of ECH/ECCD launcher for JT-60SA

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Development of a launcher for high-power (7 MW), long-pulse (100 s), and dual-frequency (110 GHz, 138 GHz) Electron Cyclotron Heating and Current Drive (ECH/ECCD) in JT-60SA is in progress. In order to realize a wide range of beam steering angles in both poloidal (~ 60°) and toroidal (~ 30°) directions and high reliability of active cooling of mirrors simultaneously, the launcher is being developed based on a linear-motion antenna concept. The antenna enables the poloidal beam steering only by a linear-motion of a small mirror while avoiding to use flexible tubes for cooling water of the movable mirrors in the vacuum vessel. Key components are a long-stroke (~ 400 mm) bellows used as a vacuum boundary enabling linear-motion of the driving shaft, and a bellows installed vertically with respect to the axis of the shaft enabling rotational motion of the shaft which results in toroidal beam steering. Recently, the bellows structure was re-designed for easy maintenance. In order to enable replacement of the bellows for toroidal beam steering, the bellows is connected by a metal O ring in the new design, which was welded in the former design. Then, a full scale mock-up of the bellows structure was newly fabricated for the cyclic test in order to confirm its reliability. Moreover, an effect of higher order mode on the diffraction loss in the antenna was quantitatively evaluated in calculation. It was shown that the truncation loss at the first mirror with a typical mirror setting (toroidal angle = 0°, poloidal angle = +20°) is increased by a factor of 1.4 by adding LP11 mode fraction of 10% in comparison with pure HE11 mode while the truncation loss rate is still lower than 1% and acceptable. The further optimization and mock-up test are planned in 2014 toward finalization of the launcher design.

Id 514

Abstract Final Nr. P3.032

22 A production of uniform negative ion beams in the JT-60 negative ion source

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In JT-60 Super Advanced for the fusion experiment, 22 A, 100 s negative ions are designed to be extracted from the world largest ion extraction area of 450 mm × 1100 mm. One of the key issues for producing such as high current beams is to improve non-uniform production of the negative ions. In order to improve the uniformity of the negative ions, a tent-shaped magnetic filter has newly been developed and tested for JT-60SA negative ion source. The original tent-shaped filter significantly improved the logitudunal uniformity of the extracted H- ion beams. The logitudinal uniform areas within a ±10 deviation of the beam intensity were improved from 45 % to 70 % of the ion extraction area. However, this improvement degrades a horizontal uniformity. For this, the unifrom areas was no more than 55 % of the total ion extraction area. In order to improve the horizontal uniformity, the filter strength has been reduced from 660 Gasus•cm to 400 Gasus•cm. This reduction improved the horizontal uniform area from 75 % to 90 % without degrading the logitudinal uniformity. This resulted in the improvement of the uniform area from 45 % of the total ion extraction areas. This improvment of the uniform area leads to the production of a 22 A H- ion beam from 450 mm x 1100 mm with a small amount increase of electron current of 10 %. The obtained beam current fulfills the requirement for JT-60SA.

Id 729

Abstract Final Nr. P3.033

Gyrotron Development at KIT: FULGOR Test Facility and Gyrotron Concepts for DEMO

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At the Karlsruhe Institute of Technology (KIT), theoretical and experimental foundations for the development of future gyrotrons for fusion applications are being laid down. Future gyrotron development might go into the following directions: • high-power gyrotrons (possible upgrades of W7-X or ITER) • higher frequency gyrotrons (more efficient current drive, DEMO) • more efficient gyrotrons with multi-stage depressed collectors • multi-frequency gyrotrons. The construction of the new FULGOR test facility (Fusion Long Pulse Gyrotron Laboratory) at KIT has started. It will comprise of a 10 MW CW power supply, a 5 MW water cooling system (extendable to 10 MW), a superconducting 10 T magnet, one or two 2 MW ECRH test loads and new control and data acquisition system for all these elements. The High Voltage DC Power Supply (HVDCPS) is versatile enough to cover a power range for gyrotrons up to 4 MW RF output power (in 2 beams), including those with multi-stage depressed collectors. It offers sufficient voltage stability to also operate gyrotrons of very high frequency (> 250 GHz) at optimum efficiency. Physical design studies towards DEMO-compatible gyrotrons (frequency above 230 GHz with a multi-frequency option, output power above 1 MW) have been started at KIT. Two different resonator designs for 237.5 GHz coaxial gyrotrons are under investigation: one optimistic design for an output power of up to 2 MW, and a more conservative design for an output power well above 1 MW. The corresponding operating modes are TE49,29 and TE49,22 respectively, chosen for multi-frequency operation. TE43,15 is a candidate-mode for a 1 MW 240 GHz conventional (non-coaxial) cavity gyrotron. It is envisaged that all the gyrotrons will have a triode magnetron injection gun. Acknowledgment This work, supported by the European Communities under the contract of association between EURATOM and KIT, was carried out within the framework of the European Fusion Development Agreement. The views expressed in this publication do not necessarily reflect the views of the European Commission.

Id 611

Abstract Final Nr. P3.034

ITER ECRH Upper Launcher Torus Diamond Window – Prototyping, Testing and Qualification

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The diamond window assembly is part of the ITER primary vacuum boundary and acts as the first tritium barrier and therefore it is classified as Safety /Protection Important Component (SIC/PIC). It consists of an ultra-low loss CVD diamond disk mounted in a system of metallic parts (copper/steel) and has to fulfil adequate transmission capability for high power mm-wave power. High power RF experiments with a 1st prototype had shown additional parasitic heating due to small gaps in the housing (effect of high frequency higher order modes localised absorption). After a design optimization directed to the mm- wave properties, the parasitic excitations of oscillations have been avoided in a 2nd prototype. This is equipped with inserted waveguide structures, which cover gaps in the metallic structure of the window housing. High power RF- measurements with a 100s pulse of 0.86 MW showed an average loss tangent of 7.1 10-6 corresponding to an increase of temperature of only 120 mK between inlet and outlet of the cooling system. The diamond window assemblies are not entirely covered by standard codes. To comply with the French safety regulations, they require therefore an ad-hoc qualification programme. In the framework of a contract with F4E, the test and qualification program for the diamond windows is being developed. Qualification tests will be executed on a (3rd) prototype, which is designed to fit to the single HELICOFLEX sealed waveguide structures of the ex-vessel mm-wave-system. The tests range from mechanical to vacuum tests up to dielectric loss measurements at low and high power. A clear definition of the testing requirements and of the acceptance criteria is necessary as well as a complete documentation of the process. Acknowledgements: This work is/was supported by Fusion for Energy (contracts No. OPE-140 and OPE-467) and by the German Ministry of Research and Education (BMBF).

Id 803

Abstract Final Nr. P3.036

Actuator management for ECRH at ASDEX Upgrade

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Electron Cyclotron Resonance Heating (ECRH) is a flexible tool to address several challenges facing next generation fusion experiments, such as non-inductive current drive, impurity regulation, sawtooth destabilisation, Neoclassical Tearing Mode (NTM) stabilisation and of course heating. Ideally, all of these tasks will be performed simultaneously using limited resources. In order to optimise the resource allocation, some form of actuator management is required. Actuator management is also necessary in view of exception handling, since gyrotrons trip and NTMs occur unexpectedly, preventing experiments from reaching their intended goals. For ASDEX Upgrade the loss of experiment time due to such exceptions is significant; for long pulse experiments (and reactors) it will be unacceptable. Unfortunately, a side-effect of the flexibility of ECRH is that one source may not be replaceable by another arbitrary source. Based on poloidal and toroidal launch angles, beams could be targeting different parts of the plasma, and have different current drive characteristics. Therefore, an algorithm has been developed, which allocates gyrotrons to specific targets in real time. The optimal allocation is determined by a cost function. Targets (central heating, q=1.5 surface etc.) are configured with values describing their importance and the amount of ECRH power required to be effective. This is balanced by the power available from each gyrotron and the actuator effort in terms of mirror movement required to reach each target from its current position. The cost function tries to use as few gyrotrons as possible, and tries to avoid frequent switching between targets. Furthermore, some gyrotrons may be unavailable or allocated to a fixed target and hence cannot be re-allocated. The algorithm is aware of these restrictions and will not try to re-allocate those gyrotrons. The submission will include a detailed explanation of the algorithm accompanied by simulation results and experimental demonstrations on ASDEX Upgrade.

Id 843

Abstract Final Nr. P3.037

The asdex upgrade parameter server

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Concepts for the configuration of plant systems and plasma control of modern devices such as ITER and W7X are based on global data structures, or "pulse schedules", which specify all physics characteristics (waveforms for controlled actuators and plasma quantities) and all technical characteristics of the plant systems (diagnostics and actuators operation settings) for a planned pulse. At ASDEX Upgrade we use different approach. Observing that although physics characteristics are frequently modified on a pulse-to-pulse basis, specifically to re-configure the highly complex and flexible discharge control system (DCS), plant system operation mainly relies on standard settings, or "basic configurations" that provide guaranteed characteristics or services and evolve according to longer term session or campaign operation schedules, AUG manages these configuration items separately. A parameter server provides a unified view on the different parameter sets and acts as the central point of access and validator for the final configuration of all plant systems and the DCS. It uses default parameters stored in a parameter database, the current plant system state and configurations, and the discharge programme (DP) as sources for the computation of the complete and consistent parameter set. The DP is an XML file containing the waveforms describing the planned evolution of the upcoming pulse and also specific static parameters. Parameters from the DP always override those from other sources and when actuator systems are affected, the parameter server will actively request a configuration change. Before the actual configuration is initiated, the parameter server executes an engineering validation of the current snapshot of parameters. The behaviour of the parameter server is largely determined by scripts (ruby, shell) and can therefore easily be modified and extended without re-compilation. We describe the functionality and architecture of the parameter server and its embedding into the control environment.

Id 841

Abstract Final Nr. P3.038

Transforming the ASDEX Upgrade Discharge Control System to a General-Purpose Plasma Control Platform

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The ASDEX Upgrade Discharge Control System DCS is a modern and mature component, originally designed to regulate and supervise ASDEX Upgrade Tokamak plasma operation. In its core DCS is based on a generic, versatile real-time software framework with a plugin architecture that allows to easily combine, modify and extend control function modules in order to tailor the system to required features and let it continuously evolve with the progress of an experimental fusion device. Due to these properties other fusion experiments like the WEST project have expressed interest in adopting DCS. For this purpose, essential parts of DCS must be unpinned from the ASDEX Upgrade environment by exposure or introduction of generalised interfaces. Re-organisation of DCS modules allows distinguishing between intrinsic framework core functions and device- specific applications. In particular, DCS must be prepared for deployment in different system environments with their own realisations for user interface, pulse schedule preparation, parameter server, time and event distribution, diagnostic and actuator systems, network communication and data archiving. The restructuring has also benefits for ASDEX Upgrade itself, as it allows also plant systems to use the DCS core framework and its outstanding features for workflow synchronisation, self-surveillance and configuration-driven functionality. The article explains the principles of the revised DCS structure, derives the necessary interface definitions and describes major steps to achieve the separation between general- purpose framework and fusion device specific components.

Id 872

Abstract Final Nr. P3.039

Model predictive control of plasma current and shape for ITER

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In a magnetically confined fusion reactor, the Plasma Current and Shape Controller (PCSC) is the component of Plasma Magnetic Control (PMC) that commands the voltages applied to the poloidal field coils, to control the coil currents and the plasma parameters, such as the plasma shape, current, and position. The PCSC acts on the system pre-stabilised by the Vertical Stabilisation controller, which is another PMC component. The challenge of PMC is to maintain the prescribed plasma shape and distance from the plasma facing components, in presence of disturbances, such as H-L transitions or ELMs, and to changes of local dynamics in different operating points. Model Predictive Control (MPC) is an established advanced process control approach in the process industry. It has gained wide industrial acceptance by facilitating a systematic approach to control of large-scale multivariable systems, with efficient handling of constraints on process variables and enabling plant optimisation. These advantages are considered beneficial for PCSC, and potentially also for other control systems of a tokamak. The main obstacle to using MPC for control of such processes is the restriction of the most relevant MPC methods to processes with relatively slow dynamics due to the long achievable sampling rates, typically needed for the on-line optimisation. However, certain simplified setups allow fast on-line computation using either analytical solutions or off-line computed multi-parametric solutions. In this work we explore the practical feasibility of using MPC for PCSC in the ITER tokamak. MPC is applied to a simulation model where PMC makes use of a combination of ohmic in-vessel coils and superconducting poloidal field coils. The results are compared with the scheme of [1] and its modification with our alternative VS scheme [2]. [1] G. Ambrosino et al., IEEE Trans. Plasma Science, 37(7), 2009, 1324-1331 [2] S. Gerkšic, G. De Tommasi, Fusion Eng. Des., 2014, http://dx.doi.org/10.1016/j.fusengdes.2013.12.034

Id 795

Abstract Final Nr. P3.040

Impact of subdivertor gas dynamics on particle recirculation in a tokamak divertor

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Efficient control of particle exhaust, fusion product removal as well as impurity and density are key issues affecting the plasma performance and the achievable burn-up fraction. The understanding of the neutral gas dynamics at the divertor-subdivertor systems in tokamaks is crucial in order to link the divertor neutral pressure and the main chamber recycling neutrals, which provides the detailed information of the particle removal process. However, the plasma conditions in a fusion device dictate the gas flow regime that can be found at the divertor-subdivertor systems. For instance, for hydrogen plasmas in ITER is foreseen an operational window of gas pressure between 1 and 10 Pa in the divertor which leads to gas flows that cover a wide range of the Knudsen number, starting with the viscous regime above the dome to the transitional and free collisional in the subdivertor. For a complete description of the neutral gas flow in the divertor-subdivertor systems, sophisticated neutral models should be implemented, which not only take into account the geometrical complexity of the ITER divertor but also describe the neutral-neutral interactions sufficiently well. The scope of this work is to investigate the impact of the subdivertor neutral gas dynamics on the particle removal and its effects on the particle recirculation phenomenon on the ITER divertor. Here, the implemented numerical approach is based on the Direct Simulation Monte Carlo method which is applied to a background plasma calculated by the fluid edge code package SOLPS. For representative divertor pressures (plasma scenarios) and different effective pumping speeds at the subdivertor, the deuterium neutral gas flow patterns and particle fluxes along the subdivertor geometry are quantified, defining a unique fingerprint in each pressure regime. The numerical results also include a complete characterization of the subdivertor gas flow: pressure profiles, flow velocity, temperature distribution, and density profiles.

Id 656

Abstract Final Nr. P3.041

Simulation of MGI efficiency for plasma energy conversion into Ar radiation in JET and implications for ITER

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Unmitigated disruption in ITER may cause significant erosion of the first wall. Understanding the plasma energy conversion into radiation during the disruption in tokamaks, mitigated with massive gas injection (MGI), is among the most important physics issues for estimation of the first wall damage in ITER. Investigation of the disruption processes and mitigation of the first wall damage with MGI in modern tokamaks is performed with the aim to extrapolate the results to the ITER conditions. Experiments performed in JET have proved that fast injection of a massive amount of noble gas can effectively mitigate the disruption, transforming both, the plasma energy and the poloidal magnetic field energy into radiation. During MGI radiation heat loads the first wall more evenly than the direct plasma impact during unmitigated disruption does. However, a series of experiments with injection of various amount of Ar in JET has revealed that the fraction of radiated energy (fRAD) over the entire disruption saturated at the level of 70-80% of total discharge energy (thermal plasma energy plus the energy of poloidal magnetic field). 2D simulations of these JET experiments have been performed using the TOKES code. Ar gas injection and the Ar plasma transport along and across the magnetic field during MGI is a 3D problem at least at pre-TQ stage. Possible difference between the 3D experiment and the 2D modeling is estimated. The results of these simulations show that during few milliseconds only electrons convert thermal energy into radiation. The ion-electron equipartition time, 10 -20 ms, is too high to cool down deuterium ions. That means the ion temperature remains almost constant during TQ and during early stage of CQ, so radiated mainly the electron heat energy and the poloidal magnetic field energy. This fact give possible explanation for the saturation of fRAD at the measured level because the equipartition time between electrons and D ions does not depend of the amount of injected Ar. Extrapolation of the fRAD saturation level measured at JET to ITER conditions has been performed.

Id 752

Abstract Final Nr. P3.042

Development of high performance control system by decentralization with reflective memory on QUEST

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Plasma control systems for tokamak plasmas are required to make control signals in real-time with simultaneously acquiring various data and calculating meaningful physical quantities. Since the physical quantities and the control signals have relationship with each other, a centralized control system is principally desirable for the grasp of these parameters. However, the computational loads on the CPU of plasma control workstation (WS) become too large to build a highly integrated control system, because it makes difficult to execute in real-time. In actual, the CPU utilization of the WS for the spherical tokamak QUEST becomes almost full. We propose to develop a decentralized control system. In this system, each control system has a reflective memory connected to each other with optical fibers, and shares various data via reflective memory. The good point of this system is to increase the CPU resource. Furthermore, the electrical insulation is ensured spontaneously. On the other hand, the synchronization accuracy between each system may become worse. The GE cPCI-5565PIORC of National Instruments Corporation is used as the reflective memory, which has 256 Mbytes memory and 170Mbyte/sec transfer rate. The most popular data type to share is double-precision real type (DBL) which needs 8 bytes to represent. The actual data read or write time is measured. Especially, within the period of 4 kHz which is the period of WS, more than 1000 to 2000 DBLs can be read or write. This means about 50 Mbytes/sec transfer rate for the one directional data sharing. For the bidirectional data sharing, each system has to repeat the read-write procedure. This would take more time. In the presentation, we will introduce the actual implementation of the reflective memory to the decentralized control system and its performance.

Id 698

Abstract Final Nr. P3.043

Implementation of a new Disruption Mitigation System into the Control System of JET

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The large vessel forces and high heat loads, which are a result of disruptions in tokamaks, are a major concern for next-generation fusion experiments such as ITER. At JET a fast valve, which has been installed originally for massive gas injection experiments (MGI), was routinely used as disruption amelioration since melting of plasma facing components occurred during unmitigated disruptions after the installation of an all-metal wall (ITER-like Wall). Despite that vessel forces could be reduced by 30% and radiated energy increased by ~2, the present system suffers from a slow response as a result of a long gas delivery time to the plasma (4ms) and additional delays caused by interlocks (up to 40ms). In preparation for high current scenario experiments and to allow for additional MGI-studies a new Disruption Mitigation System (DMS) has been installed. The new fast eddy-current valve can inject five times more gas than the old DMS. In parallel the triggering system has been upgraded to achieve a faster response of the DMS. The new system 1. is expected to lead to a further reduction of vessel forces and heat loads during disruptions, 2. is suitable for runaway beam suppression experiments and 3. will allow comparison of fuelling efficiency as well as radiation asymmetries. As part of the design a hazard and operability study has been carried out, which resulted into a number of safeguards (e.g. engineered interlocks). The execution of the fast valve is integrated into the JET-real time system (RTS) and various sensors are used for the trigger. In parallel the system is continuously providing a status signal to the RTS, which will allow premature termination of the pulse in case the DMV fails. In this paper the control of the new DMS will be described and first results from its operation will be presented.

Id 815

Abstract Final Nr. P3.044

Feedback Control of Plasma Density and Heating Power for Steady State Operation in LHD

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Steady state operation is one of the most important issue for generation suitable and economically in the fusion reactor. In the plasma operation using heating devices and fuel supplies, the control system which contains the real-time monitor of the plasma and devices, and feedback control is needed for stable operation. In the steady state operation in the experimental device, stable heating power supply and gas puffing system for sustain a constant plasma density are necessary. In the experiment of the Large Helical device (LHD), ion cyclotron range of frequency (ICRF) minority heating is used for sustain the long pulse high performance plasma. There are two important tasks for the control system. First one is the sustain the plasma density and temperature by control the ICH injection power and the gas supply. The gas supply is controlled based on the Propotional Integral Derivative (PID) of the electron density measured by the FIR interferometer. When the density increase event occurs due to the ionization of impurities or additional gas from the wall, we sustain the target density by increasing the ICH injection power by the system. Second one is the restraint the ICH system for stable and safety operation. Stabilization of the ICH injection power is important in sustaining the plasma density stable as described above. In order to stabilization, when some unwilling event such as increasing the ICH power reflection occured to the heating devices, the immediate shot down and restart are important by the fast interlock system. In the developed system, many kind of interlocks are prepared for the safety operation and their response time is setting based on the oscillation of the plasma or measurement noise. Using the developed system, we achieved the stable operation up to 47 minutes with electron density of more than 1x1019.

Id 1018

Abstract Final Nr. P3.045

First assessment of microwave diagnostics for DEMO

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Microwave diagnostics, like reflectometry and ECE, with their need for reduced access, front-end robustness, space coverage and spatial resolution are strong candidates to provide DEMO with measurements of electron density and temperature profiles and their associated fluctuations. To evaluate the microwave accessibility to the plasma several DEMO1 scenarios were analyzed. O- mode reflectometry can cover the plasma from edge to core both from high and low field side with a set of frequencies ranging from 18–110 GHz. A low gradient at the plasma core can be a problem that could be magnified due to relativistic effects (Te2 keV). For plasma position and shape control several poloidal views should be used with O-mode reflectometers probing the scrape-off layer. X-mode upper cutoff can be used to probe from the edge to core with frequencies ranging from 140–250 GHz. Due to relativistic effects profile reconstruction depends on the knowledge of local Te. There could also be conditions where large plasma regions are not accessible due to hallow or no gradient profile induced by peaked Te profiles. For ECE some 40 channels ranging from 280 to 365 GHz using X-mode electron cyclotron second harmonic will give a spatial resolution of about 6 cm very similar to what it is foreseen for ITER. Using Finite-Difference Time-Domain (FDTD) full-wave codes, synthetic O- and X-mode reflectometry diagnostics were implemented and a first assessment of the response of the different lines of view. Some of them are particularly challenging since will be facing the plasma not at midplane but obliquely viewing a plasma which presents a poloidal density divergence and curvature, both adverse conditions for profile measurements. The presence of a poloidal density gradient makes the distance between the mouth of the horn and the cut-off (turning point) difficult to evaluate. It is of major interest to understand the impact of such topology in the reconstruction of density profiles and plan for error mitigation procedures.

Id 319

Abstract Final Nr. P3.046

FPGA real-time spectrum code for gamma-ray spectroscopy diagnostics

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Gamma-ray diagnostics are considered of crucial importance for understanding the plasma behaviour of next fusion devices. Among other physical phenomena, gamma-ray spectra can provide information about the fusion reaction rate and the fast ions temperature and confinement, indicators of how close we are from reaching self-sustained burning plasma. Accordingly, dedicated gamma-ray diagnostics are currently installed at the Joint European Torus (JET). The 2D gamma-ray profile monitor is one of these diagnostics, equipped with an Advanced Telecommunications Computing Architecture (ATCA) Data Acquisition (DAQ) system, capable of digitizing gamma-ray signals from the 19 photodiode detectors. The DAQ system includes Field Programmable Gate Array (FPGA) devices, with embedded processing algorithms. These algorithms are responsible for processing the gamma-ray signals acquired from each detector in real-time, and for periodically streaming the corresponding energy values to the DAQ host. The diagnostic performance is presently limited by the detectors slow response. Therefore, to cope with higher count rates (few MHz) expected during next deuterium-tritium experiments, the detectors replacement is foreseen. However, for higher count rates it is unfeasible to stream periodically all the energy values without loss, as currently provided. As so, a new algorithm was designed, capable to produce real-time spectra in the FPGA from the processed energy values. The spectra will be periodically streamed, instead of energy values, ensuring no data losses. Consequently, the streaming data can be used for control purposes, as demanded by next fusion experiments with long plasma discharges of high energy/count rate content. This work describes the real-time spectrum code developed for DAQ system FPGAs along with the attained results. It was concluded that the spectrum code is also suitable for implementation in other spectroscopy diagnostics, whenever real-time spectra are required.

Id 243

Abstract Final Nr. P3.047

ATCA Shelf Manager EPICS Device Support for ITER CODAC Core System

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The ATCA (Advanced Telecommunications Computing Architecture) with extensions for Physics was selected as one of the fast controller instrumentation standards for ITER diagnostics. The ITER Fast Plant System Controller (FPSC) prototype, based on this technology, contributed to the ATCA test and standardization procedures fulfilling the ITER requirements, which demonstrated its compliance with complex applications for Physics. According to the ATCA specifications, the Shelf Manager (ShM) is a key element for the ATCA operation responsible for system management and monitoring, as well as for the implementation of high availability, enhancing the use of these systems for long pulse operation. Using the Intelligent Platform Management Controllers (IPMC) it monitors the health of the system, retrieves inventory information and controls the performances of Field Replaceable Units (FRUs), ensuring the correct operation of the system and also the installed boards. The ShM controller can be accessed using a connected terminal or several protocols over Ethernet like RMCP (IPMI over TCP/IP), HTTP or SNMP. The ITER CODAC Core System (CCS) is responsible for the plant Instrumentation and Control (I&C) supervising and monitoring. This system uses the Enhanced Physics and Industrial Control System (EPICS) Channel Access (CA) protocol as the interface with the Plant Operation Network (PON). This paper presents a generic EPICS device support implemented according to the Nominal Device Support (NDS) specification for scalability and easy configuration. The device support uses the HTTP interface for the ATCA ShM integration into the ITER CCS. Both HTTP server and sensors/actuators definition can be configured using the EPICS Database File and the Input / Output Controller (IOC) initialization file. The EPICS device support running in an IOC provides Process Variables (PV) to the PON network with the system information. These PVs can be used by all CA clients like EPICS user interface clients, alarm systems and archive systems to report problems, show the system environment in the control room and store data for future analysis. Operation with redundant ATCA ShMs and device support scalability tests were performed and the results are presented.

Id 602

Abstract Final Nr. P3.048

ATCA/MTCA Data Acquisition System for Advanced Fusion Experiments

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An in-house high-frequency Data AcQuisition (DAQ) system, based on the Advanced Telecommunication Computing Architecture (ATCA), was developed aiming at supporting large-scale physics experiments. The accumulated work carried out on the various fusion experiments revealed a set of desirable features to be introduced in a new fast DAQ system directed to the International Thermonuclear Experimental Reactor (ITER) demanding diagnostics, such as Thomson scattering, microwave reflectometry or for the plasma position reflectometer. Giving relevance to time-to-market, where system upgrade can be quickly implemented, a new high-frequency DAQ system was built with a modular architecture. The system complies with ITER solution preference for fast Input/Output (I/O) instrumentation systems which separates I/O chassis from CPU/Network chassis and it is based on an Advanced Mezzanine Card (AMC) built upon a commercial Xilinx®; Kintex-7™ Field Programmable Gate Array (FPGA) module. The commercial FPGA module granted the prototype production time reduction. Moreover, this AMC module supports an FPGA Mezzanine Card (FMC) for I/O expansion. This new architecture provides hardware modularity and reconfigurability which are some of the system major advantages. The paper is focused on a full-size AMC module designed to cope with the PICMG AMC.0/AMC.1 R2.0 and VITA 57.1 FMC standards and on a dual-channel high-resolution (12-bit), high frequency (1.6 GHz), low-noise Analog-to-Digital converter FMC module (FMC-AD). The AMC module implements complex I/O interfaces, supports high-speed data transfer protocols to the system host and performs in-stream data processing allowing long period system operation without data loss. Besides the high sampling rate of the FMC-AD module, signal conditioning circuitry, where Alternate Current (AC) or Direct Current (DC) input coupling can be implemented (depending on the application), appears as the main focus of this module, as this circuitry bridges the lack of DC input coupling of analog-to-digital commercial modules.

Id 809

Abstract Final Nr. P3.049

Precision time protocol synchronization support hardware for iter tcn compliancy

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An in-house, ATCA-based control and data acquisition subsystem has been developed, aiming for the ITER Fast Plant System Controller (FPSC). Timing and synchronization for the ATCA modules links to ITER CODAC through the Timing Communication Network (TCN), which uses IEEE 1588-2008 Precision Time Protocol (PTP) to synchronize devices and to provide the correspondent grandmaster clocks. The TCN infrastructure was tested for an RMS jitter under the limit of 50 ns. IPFN's hardware, namely the ATCA-PTSW-AMC4 hub-module, which is in charge of timing and synchronization distribution for all the nodes, shall also comply with this jitter limit requirement. This paper describes relevant hardware upgrade applied to the ATCA-PTSW-AMC4 hardware to comply with these requirements, in particular the inclusion of an add-on mezzanine module on the Rear Transition Module (RTM). This add-on is based on a commercial FPGA-based module from Trenz Electronic, allowing the implementation of the PTP protocol service linking to TCN, as well as customized functions for the ATCA-based subsystem timing. The programmability of these devices is one immediate advantage, allowing the quick implementation of the desired features and upgrades. The paper also presents results obtained after the tests at the ITER facility and, finally, indicates further developments to the system, in order to fulfil ITER's requirements.

Id 595

Abstract Final Nr. P3.050

PCI Express Hotplug Implementation for ATCA based Instrumentation

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This paper describes a PCI Express hotplug capability feature implementation for the ITER FPSC (Fast Plant System Controller) project ATCA specification compliant. Hotplug capability feature provides board insertion and removal from a healthy running platform without causing system damages. Control and data acquisition cards are typically inserted and removed from a system for: (i) fault-condition repair; (ii) hardware malfunction replacement; (iii) firmware updates or upgrades; (iv) hardware reconfiguration without requiring an entire system shutdown, providing to control operators the ability to isolate card(s) in the event of a failure occurrence. ATCA specification key features such as high reliability and high availability for CODAC (Control Data Acquisition and Communication) systems strongly benefits from hotplug capabilities taking advantage from present Linux operating systems and corresponding kernel upgrades with built-in mechanisms and embedded software modules, fully hotplug-supported. Implementation of hotplug feature in the ATCA-based ITER FPSC system provides described capabilities to all platform data acquisition and timing hardware modules enabling the implementation of a fast replacement strategy of damaged boards hence leading to a significant reduction of systems downtime. This feature also aims to provide to the selected platform operating system ability to automatically load and unload cards' corresponding device drivers in compliance with ITER requirements and specifications.

Id 558

Abstract Final Nr. P3.051

Phase modulated ADC module for long term numerical integration of magnetic signals

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Forthcoming advanced fusion experiments like the superconductor based ITER and W7-X will target near steady state operation. Long term and low drift integration of magnetic signals is required for safe and high-performance operation. A phase modulated (chopper) ADC module (one channel and two channels versions) was designed targeting the digital integration, over periods of time higher than 1 hour. The modules include differential inputs for minimizing EMI and earth-loop effects on the signal, with settable dynamic ranges, impedances and analogue filters cut-off frequencies, the 18-bit resolution @ 2MSPS digitizer and a 1 kV galvanic isolation barrier on the digital domain. Up to 24 ADC modules (48 channels) can be installed on specially developed ATCA carrier board that contains a VIRTEX-6 field programmable gate array (FPGA). Algorithms for online offset estimation/correction and numerical integration were developed on HDL to run on FPGA. Finally, the integrated signals are transferred to the Host PC through low latency (< 10us) cabled PCIe interconnect enabling real-time data processing or transmission. A series of performance tests were carried in diverse environmental conditions (Temperature variation, external magnetic field) have shown integration drifts lower than 0.2 mVs for 1 hour (.06 uVs/s) and that the reliable operation modules is compatible with external DC magnetic field up to 12 mT.

Id 824

Abstract Final Nr. P3.052

Progress on Diagnostics Integration in ITER Equatorial Ports #11 and #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 diagnostics on ITER have entered the preliminary design phase following the handover of several work-streams to the Domestic Agencies, it is important to provide the diagnostic 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 Diagnostic Shield Module concept as well as in the design of maintainable vacuum extension connections at vacuum boundary and in-vessel electrical connections. The Diagnostic Shield Module concept must meet two competing requirements: weight limitation and nuclear shielding. A solution using boron carbide integrated in a stainless steel structure which houses the diagnostics components has been developed and analysed. The design of vacuum extension disconnections is highly constrained by requirements for environment protection from contamination and tritium leakage as well as with personal safety requirements linked to their high activation. A remote-handable connector concept has been studied based on the use of several shutters and additional tritium pumping. Finally a remote handling compatible generic concept of the Electrical Assembly that can be adapted for each of the Diagnostic Shield Module of the Equatorial Port Plugs has been defined. Two particular ports, the equatorial port 11 and the equatorial port 17, are used to illustrate the concepts. Engineering solutions will be addressed and discussed.

Id 188

Abstract Final Nr. P3.053

Maturity Assessment of ITER Diagnostics Plant Instrumentation and Control Design

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ITER requires extensive diagnostics to meet the demands for machine operation, protection, plasma control and physics studies. The interfaces between plant instrumentation and control (I&C) and the central control system follow mandatory rules described in the Plant Control Design Handbook (PCDH), while the design strategy for PCDH compliant plant I&C is covered in its guidelines and supported by hardware catalogues. During preliminary and final design review as well as factory and site acceptance testing it is therefore important to determine the maturity of the I&C design and its implementation to ensure its compliance with the PCDH and the diagnostics performance requirements. In the design phase the maturity evaluation of the documentation covers the following topics: (1) operation principles, (2) functional analysis including all variables and their attributes, (3) hardware architecture with all signal connections, (4) cubicle layout and (5) plant system operating states. For acceptance testing the focus is on (1) test plans, (2) test reports and (3) operation and maintenance manual. All diagrams and their attributes are documented using the system engineering tool Enterprise Architect. Evaluation criteria in every life-cycle phase are the coverage, completeness, level of detail, and quality of contents for all topics. For the purpose of design reviews the maturity of the documentation is determined through comparison of each of the I&C deliverables ítems listed above with existing and reference designs and implementations (diagnostics use cases) as well as expert judgement. This paper presents the maturity assessment process for the ITER diagnostics plant I&C currently under development and elaborates on the metrics used.

Id 290

Abstract Final Nr. P3.054

Design and development activities for in-vessel and in-port components of ITER microwave diagnostics

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ITER tokamak will be operating with 5 microwave diagnostic systems: EC Emission measurement (ECE), Low Field Side Reflectometry (LFS-R), High Field Side Reflectometry (HFS-R), Plasma Position Reflectometry (PPR) and Collective Thomson Scattering (CTS). While they rely on different physics, they share a common need: transmitting low and high power microwave in the range of 12 to 1000 GHz (different bandwidths for different diagnostics) between the plasma and a diagnostic area tens of meters away. Microwaves are guided between the vacuum vessel and the emission/detection equipment using oversized waveguides. To pass the vacuum barriers, vacuum-tight, safety-important windows are used. On top of their role as vacuum container, they also have to contain tritium in case of accidental events (severe disruptions, earthquakes, fire…). Most of them are in fact a set of double windows with monitored interspace; others are paired with a vacuum shutter. They are critical in term of safety but also for the good performances of the diagnostic: they have to be designed to minimise the microwave reflection while keeping a robust and safe approach to meet the requirements of a radiological barrier. In the vacuum vessel, waveguides are used to route microwaves between windows and antennas. Space available being tight, the route has to be carefully designed in order not to jeopardize measurements made through these small waveguides. They also have to accommodate the requirements of a tokamak environment: volumetric heat loads due to neutron flux, electro-magnetic loads and seismic loads. Finally, some in-port plug antennas, being close to the plasma, may be actively water cooled. Others are placed in restricted position and have been design to accommodate the small space allocation while keeping good broadband microwave performances. The designs proposed for vacuum windows, in-vessel waveguides and antennas will be presented together with the development activities needed to finalise this work.

Id 341

Abstract Final Nr. P3.055

Final Design of the Generic Equatorial Port Plug Structure for ITER diagnostic systems

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The ITER Project is an international effort aimed at demonstrating the scientific and technological feasibility of nuclear fusion energy. A large number of different types of diagnostic equipment peer into the ITER vacuum vessel from many different vantage points. The focus of the present paper is one generic location known as the equatorial port plugs. The Diagnostic Generic Equatorial Port Plug (GEPP) is designed to be common to all equatorial port-based diagnostic systems. It is designed to survive throughout the lifetime of ITER for 20 years, 30000 discharges, and 3000 disruptions. The EPP structure dimensions (without Diagnostic First Walls and Diagnostic Shield Modules) are L2662 x W1938 x H2390 mm3. The length of the fully integrated EPP is 3174 mm. The weight of the EPP structure is about 15 t, whereas the total weight of the integrated EPP may be up to 45 t. The EPP structure provides a flexible platform for a variety of diagnostics. The Diagnostic Shield Module assemblies, or drawers, allow a modular approach with respect to diagnostic integration and maintenance. In the nuclear phase of ITER operations, they will be remotely inserted into the EPP structure in the Hot Cell Facility. The port plug structure must also contribute to the nuclear shielding, or plugging, of the port and further contain circulated water to allow cooling during operation and heating during bake-out. The Final Design of the GEPP has successfully passed in late 2013 and is now heading towards manufacturing. The final design of the GEPP includes interfaces, manufacturing, R&D, operation and maintenance, load cases and analysis of failure modes.

Id 456

Abstract Final Nr. P3.056

Performance of the ITER interlocks slow architecture prototype

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The Interlock Control System (ICS) of ITER is in charge of implementing the investment protection functions. It is composed of the Central Interlock System (CIS) and around thirty Plant Interlock Systems (PIS). Among these, an estimated number of at least twenty two PIS will interface the CIS through the so-called slow architecture based on PLC technology. Several tests were conducted on a slow prototyping platform in order to validate the selected technology in terms of performance and to assess its limitations. These test results will help to define the final ICS architecture.

Id 154

Abstract Final Nr. P3.057

An Overview of the ITER Cabling Network and Cable Database Management

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ITER is a large nuclear fusion facility, where systems with varying functional and physical characteristics coexist. This makes the cabling requirements diverse too. The cabling network design for ITER demands extensive integration of many systems interfaces. This process starts by collating cable requirement data to form a giant cable database with scope for approximately 120,000 cables in total. 60,000 cables have already been registered within the ITER cable database. Ultimately the cabling installation will be approximately twice the size of one required for a 3rd generation nuclear fission power plant, with cables spanning more than 9,000 kilometres installed within approximately 200 kilometres of cable trays. Considering the differing cabling requirements, the raceway network and cables shall be routed in a prescribed manner to comply with French Nuclear Safety standards. In order to perform the cable engineering design, various application software tools are being used such as: “SEE Systems Design” (SSD) by IGE-XAOa for the 2D schematic (cabling diagrams). SSD performs a dual role for the entire ITER cabling network, it captures the 2D cable schematic design and at the same time, registers each cable type and serves as the Central Repository for all ITER plant system cables that comprise the giant cable database. 3D modelling of the cable trays and supports has been performed in CATIAb in conjunction with models of ITER buildings and system components. Integration of the 2D and 3D design has been implemented by KCMSc; Bespoke application software developed by KEPCO E&C for the entire ITER cabling network. This paper describes an overview of the ITER cabling network, the cable engineering design workflow, development and management of the cable database and the interfaces between the software tools (SSD, CATIA, and KCMS). a IGE-XAO, A company group which produces Electrical CAD software packages. b CATIA, Computer Aided Three-dimensional Interactive Application, software suite developed by Dassault Systèmes c KCMS, KEPCO E&C Cable Management System, a software which integrates both SSD(2D) and CATIA(3D) data.

Id 740

Abstract Final Nr. P3.058

ITER In-Vessel Viewing System engineering analysis

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The In Vessel Viewing System (IVVS) will be one of the essential machine diagnostic systems at ITER to provide information about the status of in-vessel and plasma facing components and to evaluate the dust inside the Vacuum Vessel. The current design consists of six scanning probes and their deployment systems, which are placed in dedicated ports at the divertor level. These units are located in resident guiding tubes 10 meters long, which allow the IVVS probes to go from their storage location to the scanning position by means of a simple straight translation. Moreover, each resident tube is supported inside the corresponding Vacuum Vessel and Cryostat port extensions, which are part of the primary confinement barrier. As the Vacuum Vessel and the Cryostat will move with respect to each other during operation (especially during baking) and during incidents and accidents (disruptions, vertical displacement events, seismic events), the structural integrity of the resident tube and the surrounding vacuum boundaries would be compromised if the required flexibility and supports are not appropriately assured. This paper focuses on the integration of the present design of the IVVS into the Vacuum-Vessel and Cryostat environment. It presents the adopted strategy to withstand the main interfacing loads without damaging the confinement barriers and the corresponding analysis supporting it.

Id 696

Abstract Final Nr. P3.059

Development of instrumentation and control systems for the ITER diagnostic systems in JADA

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Japan Domestic Agency (JADA) is responsible for six diagnostic systems in the ITER project. We have successfully developed an instrumentation and control (I&C) system for the JADA thermocouple measurement system which includes a supervisory function, a sequencing management function, and a data acquisition function. The supervisory function was implemented using Experimental Physics and Industrial Control System (EPICS) and state notation language (SNL). This function manages internal operations for measurement such as health checks of sensors, configuration of measurement parameters, and consistency checks between measurement parameters. We developed a conversion tool to convert operational flowcharts to EPICS records and a sequencing management function using the Python language. The records generated by the conversion tool are used to trigger each operation step and to indicate the progress of the sequence. The sequencing management function coordinates the execution of operation steps by monitoring changes in record values. It was designed so that the relationship between the records and steps is determined automatically according to the flowcharts as much as possible. The data acquisition function is conducted by the supervisory function and the sequencing management function. We validated the performance of the I&C system for the thermocouple measurement system, and are continuing the development of even more complex I&C systems for other JADA diagnostic systems.

Id 377

Abstract Final Nr. P3.060

Development of magnetic sensors 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. Therefore, we prepare RWM control system for JT-60SA, in order to achieve steady-state high beta plasma and also to clarify stabilization mechanism of RWM. For RWM stabilization, error field correction is necessary because error field destabilize RWM by reducing plasma rotation and resonant field amplification. Therefore, we will install coils in the vacuum vessel for RWM control, error field correction and also for ELM control. For these controls, we need the 3D information of magnetic configuration. Therefore, we have developed the biaxial magnetic sensor. Also for basic information of JT-60SA plasma, we developed the magnetic sensors, e.g. one-turn loop, Rogowski coils, diamagnetic loop and saddle coils. Because the length of the vacuum vessel in the poloidal direction of JT-60SA is almost twice longer than that of JT-60U, the length of the Rogowski coil and the diamagnetic loop on JT-60SA, which are for the measurement of plasma current and plasma stored energy, respectively, are also twice longer than those on JT-60U. Construction and installation of these are much more difficult for JT-60SA. Therefore, we have developed these with new design. We will report the design and specification of the magnetic sensors in JT-60SA from engineering and physics aspects.

Id 667

Abstract Final Nr. P3.061

Neutronic Analysis for Detail Desing of Optical System of the Edge Thomsong Scattering System for ITER

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Neutronic analysis is very important to optimize design of diagnostics system of ITER. In this paper, results of neutronic analysis applied to the ITER Edge Thomson Scattering (ETS) system, which is procured by Japan Domestic Agency, are presented. The ETS system measures electron temperatures and density profiles of edge plasmas (r/a > 0.85). The ETS system mainly consists of the laser injection and the collection optics systems installed in the equatorial (EQ) port plug #10. The detail design of those systems should be considered from the viewpoint of not only measurement accuracy but also neutron shield, because the laser injection system must make direct penetration through the port plug and the collection optics system requires large aperture in the front of the port plug to collect scattered light of the injected beam effectively. In order to shield neutrons efficiently, it is necessary that the optical system has an effective labyrinthine structure and materials of radiation shield must be optimized. In this study, nerutronic analysis with MCNP-5 was performed to optimize the design of the ETS system. For the collection optics system, deep labyrinthine structure was designed while the measurement capabilities were maintained. Further, effects on radiation shield due to differences in materials were assessed. The results indicated that neutron flux in interspace region behind the port plug could be reduced by more than half order by utilizing boron carbide (B4C) as substitute for conventional shield using Stainless and Water. For the laser injection system, the additional shield around injection path was designed in the inter space region. Further, the laser injection path is bent twice by optical mirrors and neutron dump is installed behind optical mirrors to reduce streaming neutrons. Neutronic analysis suggested that those updated design could also reduce neutron flux by ~ 60 %.

Id 981

Abstract Final Nr. P3.062

The EFDA Goal Oriented Training Program GOT-4 Diagnostic Techniques

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Abstract The construction of a new Tokamak like ITER requires deep knowledge in different fields of engineering and physics. In order to provide such knowledge also for future development of fusion science, it is intended to train early stage engineers and physicists during 3 years. The training takes place within a large collaborative group of research centers located in Europe. The structure of the training is organized in work packages (WP), dealing with the diagnostics that will be necessary to be integrated inside the reactor to provide safe operation and avoid hazards to the plant workers. The partners are the KIT in Karlsruhe (WP1), CEA in Cadarache (WP2a and WP2b), ENEA in Frascati (WP3), CNR-IFP in Milano (WP4), FZJ in Jülich (WP5), CIEMAT in Madrid (WP7) and HAS in Budapest (WP9). The various work packages deal with different kind of diagnostics. KIT studies the upper port plug diagnostics, with a major focus on arc, tritium and microwave detection at the diamond window assembly. CEA deals with ITER Equatorial Visible/Infra-Red Wide Angle Viewing System (WAVS) and the calorimeters required for a modern fusion plant. ENEA is developing the Radial Neutron Camera, while CNR-IFP is responsible for the CNESM neutron diagnostic for the neutral beam injector. FZJ focuses its work on the optical diagnostics (CXRS), while at CIEMAT the main topic are the neutronic studies and port plug integration of the ITER plasma position reflectometer and WAVS. HAS is finally responsible for the beam emission spectroscopy and its integration in the port plugs. The aim of this paper is to briefly present the work produced so far by the various partners and to provide also a view on the future development. "This work, supported by the European Communities under the contract of Association between EURATOM and the involved associations (see above), was carried out within the framework of the European Fusion Development Agreement (GOT4-DIAG). The views and opinions expressed herein do not necessarily reflect those of the European Commission."

Id 180

Abstract Final Nr. P3.063

Neutronics analysis for ITER cable looms

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Diagnostics systems are essential components of the ITER tokamak to fulfill its aims. Many of them are composed with magnetic sensors and loops, providing measurement parameters for plasma control and operation at ITER. These systems are connected by cables distributed all around the ITER Vacuum Vessel (VV) and divertor cassettes. The cables are collected in cable looms, each loom could house up to ~40 cables as a well compacted assembly inside the very limited space, such as the gap between the blanket modules and VV or grooves in the divertor cassette bodies. The cable looms must withstand harsh radiation environment of D-T 14 MeV plasma neutron source and secondary photons. Therefore, providing neutronics analysis is important for the cable looms' development. Several neutronics characteristics were analysed in this paper. Most important is to reduce the temperature variation of the cables along the entire route inside the VV, by achieving good thermal contact with the water cooled VV structure. For this purpose, nuclear heating has been calculated with neutron and photon heat depositions for several materials of the concept cable loom design: copper, steel, cable mineral insulator, and INCONEL Alloy 718. To estimate radiation effects on the cable looms, the neutron fluence in two energy groups: fast (E>0.1 MeV) and thermal-epithermal energies was calculated. Neutron damage of the cable mineral insulator was estimated with the standard Norgett–Robinson–Torrens (NRT) model in units of displacement per atom (dpa). Neutron induced nuclear reactions also lead to transmutation of the chemical elements of the original cable looms, e.g. pure Cu metal transmutes to alloy Cu with Ni, Zn, while Au transmutes to allow Au with Hg. These neutronics problems were investigated by means of thorough 3D radiation transport analysis with the Monte Carlo MCNP5 code and activation inventory calculations with the FISPACT code.

Id 440

Abstract Final Nr. P3.064

Experimental results and validation of a method to reconstruct forces on the ITER Test Blanket Modules

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The Test Blanket Modules (TBMs), which will be located inside the equatorial port plugs of ITER, are connected to the shield by an attachment system. One of the most demanding loading conditions will be high electromagnetic forces acting on the TBM box during operation. In order to estimate these forces during operation in ITER, a force reconstruction method is developed. The reconstruction is based on measurements of strain sensors on the attachment system. Finally, the force estimates can be used to validate the results obtained by FEM software. A testing device with a modular setup has been built to support the development of the method. It is able to apply different loading conditions to a corresponding mock-up, representing a TBM with attachment system. The attachment system is equipped with a set of strain sensors. Two force reconstruction methods, the Augmented Kalman Filter and an optimization algorithm, have been selected and adapted to estimate the excitation forces. Different test cases have been defined to represent a complete set of possible excitations of the systems. This paper demonstrates the feasibility of the application of the method to reconstruct forces on the TBM structure. This is supported by results of the force reconstruction with experimentally obtained strain recordings as well as with simulated strain data. As the algorithms are based on a model of the system, the simulated strain recordings are used to show the impact of modelling errors on the accuracy of the estimated forces.

Id 965

Abstract Final Nr. P3.065

Design and qualification of a MGy tolerant front-end chip in 65 nm CMOS for the read-out of remotely operated sensors and actuators during maintenance in ITER

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The recently introduced ITER policy on electronics exposed to nuclear radiation gives priority to radiation mitigation (shielding, relocation, replacement, passive functions). It defines critical electronics as electronics which participates in safety or investment protection functions, or whose failure would prevent machine operation. It does not authorize critical electronics to be exposed to radiation. Non-critical electronics can be exposed to radiation if duly radiation-qualified, e.g. radiation-hard electronics, or radiation-qualified COTS electronics, or radiation-qualified custom electronics, including ASICs designed with radiation tolerant layout and architecture in properly selected sub-micron CMOS technology. ITER remote handling equipment operates in high radiation environments in and around the ITER machine. In general, radiation sensitive electronic boards are centralized in control cubicles located in radiation-free areas, and are connected to actuators and sensors through long stretches of cables. However, we forecast the need of some non-critical front-end electronics that will have to be located close to those actuators and sensors. In particular, for Divertor remote handling, it is estimated that these components will face gamma radiation up to 300 Gy/h and a total dose of 1 MGy. In order to comply with the ITER relevant environmental constraints, to implement a timely procurement for a full system integration and also to benefit from economies of scale, a generic system on chip has therefore been designed and processed for qualification. The System-on-chip (SoC) consists of 8-channel programmable gain instrumentation amplifiers, 8-channel delta-sigma 16-bit ADCs, a multiplexer, an on-chip silicon temperature sensor, a voltage reference and a clock reference. It allows the read-out and multiplexing of temperature sensors, strain-gauge based pressure sensors, resolvers and position sensors (e.g LVDT) exposed to a cumulated dose exceeding MGy levels. The qualification process and results obtained with this SoC manufactured in 65 nm CMOS technology will be presented in the final paper.

Id 908

Abstract Final Nr. P3.066

Mobile robot for the inspecting in vacuum vessel of ITER

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The inspection of the vacuum vessel (VV) of the ITER from inside or outside is very difficult because of various constraints, such as non-magnet effect material, high temperature, constrained space, radiation etc. Therefore, special robots are required for the inspecting tasks. In the assembly of the VV, the NTD test carries out after welding, and a film need to be brought to the other side of the welded wall to capture the X-ray and detect the defect-welded points. As space is very limited, the current solution is to build up a tracker on the back of the wall, in order to guide a robot to the welding gap. However, it costs very much and affects the function of the VV, due to the extra tracker on the back of the wall. This paper presents a special mobile robot, which can freely move on the non-magnet material surface in any position within a confined space. The robot is designed base on the principle of vortex. The configuration and control system are presented in the paper. The robot is able to carry out the NTD test for detecting the welding and inspecting the first-wall surface.

Id 737

Abstract Final Nr. P3.067

Polarimetry data inversion in conditions of tokamak plasma: model based tomography concept

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Model based plasma tomography is studied which fits a hypothetical multi-parameter plasma model to polarimetry and interferometry experimental data. Fitting procedure implies minimization of the error function, defined as a sum of squared differences between theoretical and empirical values. Minimization procedure for the function is suggested to be performed using the gradient method. Contrary to traditional tomography, which deals exclusively with observational data, model-based tomography (MBT) operates also with reasonable model of inhomogeneous plasma distribution and verifies which profile of a given class better fits experimental data. Model based tomography (MBT) restricts itself by definite class of models for instance power series, Fourier expansion etc. The basic equations of MBT are presented which generalize the equations of model based procedure of polarimetric data inversion in the case of joint polarimetry-interferometry data.

Id 269

Abstract Final Nr. P3.068

Polarization properties of a corner-cube retroreflector

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The cube-corner retroreflector, an optical element consisting of three orthogonal mirrors, reflects incident beam in the exactly opposite direction. The change in the polarization state of the infrared beam, reflected from CCR constructed of three metal surfaces with complex reflexion coefficient, is discussed theoretically with ray tracing and the Jones matrices formalism. It is found that the final polarization state is modified according to the angle of incidence, the ray path through the reftrorfletor as well as the beam wavelength and its initial polarization. The main point is that under the specified conditions polarization changes are minor, what is essential for the CCR practical application at tokamak polarimetric systems.

Id 268

Abstract Final Nr. P3.069

Present status of the new Power Supply Systems of JT-60SA procured by EU

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JT60SA, the superconducting tokamak under construction in Japan, will be equipped with a mix of new and reused Power Supplies (PS). Most of the new PS are procured by European Voluntary Contributors under the framework of Broader Approach agreement between F4E and JAEA. For the toroidal circuit, the 6 pulses ac/dc converter will be procured by CEA. It is rated 25.7kA and 80V dc, and will work in steady state condition. For the poloidal circuits the procurement of ten new ac/dc converters, rated ±20kA and about ±1kV is shared between CEA and ENEA. They are 24 pulses four quadrant converters, with back to back thyristor bridges. In order to allow a smooth crossing of zero current, a particular control mode with circulating current operation will be implemented. The fast control of plasma position is obtained by two in-vessel coils supplied with independent thyristor converters procured by ENEA, each one rated ± 5kA and ±1kV. Plasma initiation requires a fast variation of current in the Central Solenoids, obtained with the insertion of a settable resistor in series to the coils. This is achieved with the operation of four Switching Network Units procured by ENEA, producing up to 5kV at the nominal current of 20 kA. The protection of superconducting magnets, both toroidal and poloidal, is assured by thirteen Quench Protection Circuits procured by Consorzio RFX, rated ±20kA and ±3.8 kV for poloidal QPCs and 25.7kA and 2.8kV for toroidal ones: in case of coil quench or other fault requiring a fast de-magnetization of the coils, each QPC inserts in the circuit a dump resistor discharging the energy stored in the coils. Finally the suppression of RWM is actively performed by 18 in-vessel coils, independently supplied by a corresponding number of IGBT inverters procured by Consorzio RFX, characterized by high dynamic performances and low latency of 50 μs, and rated for 300A and 240V. The present status of the aforementioned PS is described in the paper: their detailed design has been completed and some systems have been already manufactured and tested.

Id 523

Abstract Final Nr. P3.070

Progress of the Engineering Analyses for the JT-60SA Toroidal Field Coils Structures

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JT-60SA is a fully superconducting tokamak presently being built at the JAEA Naka site, in the framework of the JA-EU Satellite Tokamak Programme under the Broader Approach (BA) Programme and JAEA's national programme. The mission of the JT-60SA project is to contribute to the early realization of fusion energy, supporting the exploitation of ITER and research towards DEMO by addressing key physics issues. The whole magnet system of JT-60SA has a cold mass of more than 670 tons, and rests entirely on the Toroidal Field (TF) magnet system, which is made of 18 D-shaped NbTi superconducting coils, each enclosed in a stainless steel casing, which are supported toroidally by a fully bolted Outer Intercoil Structure (OIS), and connected to one another by complex bolted and pinned Inner Intercoil Structures (IIS). Due to the demanding electromagnetic loads experienced by the tokamak magnet system during operation, together with the tight space requirements around the coils, the intercoil structures have to transfer massive shear and normal forces, in excess of several MN, while providing enough flexibility to avoid unnecessary internal reactions to the immense centripetal and wedging forces at the magnet centre vault. This paper illustrates the engineering analyses carried out to define the requirements for the coil casings and intercoil structures, to assist the development of the technological solutions in view of the manufacturing of the components, and to predict the behaviour of the components from the assembly up to off-normal and fault conditions during operation. The latest results obtained from the global electromagnetic and structural analyses, briefly presented in the paper, show that the magnet system is suitable to perform the intended number of cycles in the operating conditions. More emphasis is given to the results of the multiple detailed analyses of the TF intercoil structures, which show that each individual component can withstand the operating loads with a comfortable safety margin.

Id 862

Abstract Final Nr. P3.071

EU ITER TF Coil: Dimensional metrology, a key player in the Double Pancake integration.

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The ITER Toroidal Field (TF) magnet system consists of 18 “D” shaped coils. Fusion for Energy (F4E), the European Domestic Agency for ITER, is responsible for the supply of 10 out the 19 TF coils (18 installed plus one spare coil). Each TF coil, about 300 t in weight, is made of a stainless steel case containing a Winding Pack (WP). Each WP comprises 7 Double Pancakes (DP), stacked together and impregnated. Each DP (approx. 14 m x 9 m) contains a stainless steel Radial Plate (RP) with machined spiral shaped grooves, in which an electrically insulated Nb_3Sn Cable In Conduit Conductor (CICC) is inserted after undergoing a wind, react and transfer process. In particular the “React” stage consists of a specific heat treatment cycle up to 650 °C. The European manufacturing of the RPs and WPs have been awarded to two different industrial partners. The 2 industrial activities are strongly linked with each other. In order to manufacture a DP, first, the conductor has to be bent onto a D-shaped double spiral trajectory, then heat treated and then it needs to be inserted in the grooves of the radial plate. This represents the most challenging manufacturing step: in order to fit inside the groove the double spiral trajectory of the conductor must match almost perfectly the trajectory of the groove, over a length above 700 m. In order to achieve this, the conductor trajectory length must be controlled with an accuracy of 1 mm over a length of 350 m while the radial plate groove has to be machined with tolerances of ± 0.2 mm over dimensions of more than 10 m. In order to succeed, it has been essential to develop a metrology process capable to control with high accuracy both the DP conductor and the RP groove trajectories. This paper reports on the work carried out on the development and qualification of the dimensional metrology used to monitor the manufacturing process of the conductor. Reference is made to the final dimensional check of the Radial Plate with a specific focus on the groove centreline length. In addition the results obtained on the one to one scaled prototype DP are described. Finally, the strategy and foreseen improvements for the production of DPs are discussed.

Id 212

Abstract Final Nr. P3.072

ITER central solenoid module fabrication program

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General Atomics (GA) is under contract to manufacture the ITER Central Solenoid Modules (CSM). The contract is managed by US ITER at Oak Ridge National Laboratory, under the sponsorship of the Department of Energy Office of Science. The contract includes the design of manufacturing processes and tooling to fabricate seven CSM (6 + 1 spare) that constitute the ITER Central Solenoid. The modules will be delivered to the ITER site during 2016–2018. GA has established a fabrication facility that combines 1,500 m2 of offices with 6,000 m2 of fabrication space. Extensive building modifications were made in anticipation of the first tooling station which arrived in February 2014. The facility utilizes several cranes with up to 35T capability to handle and move delivered conductor and wound pancakes during the winding, joint and lead fabrication, and stacking operations. The assembled 110 ton coil is moved via a self-propelled air bearing cart designed specifically for this application. Critical systems have been designed, built and initial testing completed for use in fabricating the CSM. The winding stations were designed, built and successfully tested at the Tauring S.p.A factory in Turin Italy. The reaction heat treatment furnace has been designed, built and tested at Seco-Warwick facility in Sweibodzin, Poland. Insulating wrapping heads were completed by Ridgway Machines Ltd of Leicester, England and shipped to GA. Several other key tooling stations for joining the coil segments, insulating the turns and vacuum pressure impregnation equipment have been designed and built by GA. Qualification of the equipment is completed with Factory Acceptance Tests at the supplier's facility, Site Acceptance Tests at GA, and lastly the production of the mockup coil prior to using the station for producing the first CS module. Work supported by UT-Battelle/ORNL under 4000103039 and DE-AC05-00OR22725.

Id 142

Abstract Final Nr. P3.073

Feasibility testing of Hybrid DC Circuit breaker for Prototype Superconducting Magnet

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Quench protection system of superconducting magnets is mainly dependent on the reliability of the DC circuit breaker. The concept of Hybrid DC circuit breaker basically comprises of a mechanical breaker in parallel with a static breaker which provides arc-less current commutation into the dump resistor for fast ramp down and extraction of inductive energy of the superconducting magnets. This scheme has been experimentally validated with series of experiments for current upto 1.5kA DC using a combination of mechanical breaker and Insulated gate bipolar transistor (IGBT) as static switch and the test results are discussed here. In non-quench state, the mechanical breaker carries the continuous DC current minimizing on-state losses. But when the quench occurs, the current is diverted from breaker to IGBT and finally IGBT commutes the current to the dump resistor where the load current (representing the energy of the superconducting magnet in actual scenario) decays exponentially. A dedicated control system has been developed for the sequential opening and closing of the IGBT and mechanical breaker in this simulated quench protection system set up. These experimental results builds confidence in scaling up of Hybrid circuit breaker concept for implementation in Protection system for Prototype fusion grade superconducting magnets carrying maximum transport current of 30kA.

Id 257

Abstract Final Nr. P3.074

Operation of SST-1 TF power supply during SST-1 Campaigns

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SST-1 TF power supply provides the DC current for the required magnetic field of the TF coil. TF power supply includes TF transformer, 12 pulse converter, VME data acquisition and control system, bus bar, water cooled cable, protection (pyro and DCCB) and measuring (shunt and DCCT)equipments, isolator and GUI software. TF power supply is operated through a GUI software built in TCL/Tk. The VME DAC system monitors the parameters, provides On/Off commands, Vref and Iref and initiates predefined reference to emergency shutdown. The emergency shutdowns are hardwired to TF power supply converter from magnet and cryo groups. During quench power supply converter opens DCCB and dump is connected in the circuit and VME system acquires Visolator, Vshunt and Ishunt. The operator can stop the process as current ramps down to zero. Operation of TF also requires monitoring of SCR temperature during high current long pulse shot, temperature and LPM of water cooled cable. Before start up of TF power supply a quench simulation is done to check readiness of protection. The operator has to continuously monitor the TF power supply during the plasma shots. TF current can be changed to new value (higher or lower to the current flat top) without ramping down to zero. This paper describes the complete operation of TF power supply pre startup process, running process, emergency and quench process, dynamic control of TF power supply, manual interruption to new value and post process as the current pulse is completed.

Id 819

Abstract Final Nr. P3.075

Assessment of instrumentation of the magnet system of W7-X

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The modular stellarator Wendelstein 7-X is currently under final stage of construction in Greifswald, Germany. The five fold symmetric magnet system consists of 50 non-planar and 20 planar superconducting coils operated at 4K. In order to ensure safe operation, the magnet system was instrumented with 542 and 30 strain gauges on the cold and room temperature (warm) part of the structure respectively, 135 cold and 30 warm distance sensors, and 88 cold contact sensors. Fast assessment of the signals during commissioning and in the first phase of operation is an important step for the preparation of long plasma pulses. The operator must be able to intervene directly to prevent overloading of critically loaded components if signals deviate too much from FE model predictions. As a first step in preparation of commissioning, the accuracy of the sensor measurements and predictions were assessed and optimized. The sensor locations and signal transformations to meaningful quantities were carefully chosen, and two independent FE models have been locally refined to improve signal prediction after mutual benchmarking. Another important step was the creation of a graphical user interface (GUI) to compare results of different simulations. With this internally developed GUI, the utilization ratio between the load case of interest relative to the design limit of each component can be calculated. The highest utilization ratio over all components determines the overall allowable scale factor for the load case under consideration. In addition, a traffic light system has been introduced highlighting each sensor in red, yellow or green to warn the operator, depending on the deviation between measurement and prediction and the overall allowable scale factor. Finally, dedicated load cases were developed for the magnet system to commission the instrumentation. The GUI proved to be an essential tool to approve these load cases.

Id 288

Abstract Final Nr. P3.076

New Drive Converter and Digital Control for the Pulsed Power Supply System of ASDEX Upgrade

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The operation of ASDEX Upgrade (AUG) relies on three large flywheel generators (EZ2, EZ3 and EZ4). These generators feed the pulsed power supply system for the magnetic confinement (high current) and additional heating (high voltage) of the plasma with stored energies up to 3750 MJ. To assure a safe and reliable operation of the 25 to 40 years old installation, fault-prone components early have to be identified. One of these week points was the water-cooled thyristor direct converter for the motor drive of generator EZ3. Excessive transient asymmetries of the output current up to 1000 A resulted in increased melting loss of the neutral slip-ring. Mal-synchronisation caused short-circuits. Besides the temporal and financial effort for repair, consequence is the danger of a serious damage to the engine and with that the longer-term shutdown of the generator. Therefore decision was made to substitute the direct converter by a new IGBT drive converter with integrated control. The new converter has to fulfil three main objectives: Proven Technology - Reduce costs and risks using a commercially available, integrated system. Power - Sufficient power to drive and speed up / brake down the generator over a wide speed range. Control - Accurate speed control and possibilities for active and reactive power management. Especially the following challenges are described in the paper: Adaptation of a converter, originally designed for wind turbines, towards a drive system for a flywheel-motor-generator. Layout of the controller and control parameters based on accurate modelling and comprehensive simulations. Further on, the paper presents the challenges to integrate a new technology into a 35 years old system. It analyses the results of measurements obtained during commissioning, compares them to the calculated design values and reports on the performance achieved during AUG plasma experiments. This project has received funding from the Euratom research and training programme 2014-2018.

Id 845

Abstract Final Nr. P3.077

Control system and DC-link supply of the inverter system BUSSARD for ASDEX Upgrade in vessel saddle coils

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For the nuclear fusion experiment ASDEX Upgrade (AUG) it is planned to assemble an inverter system to individually feed the 16 in-vessel saddle coils. In a first step, four inverters to power groups with up to 4 serial connected coils are assembled. The feeding of the common DC-link is done by an existing current converter. The connection to this converter, the monitoring of the new inverters and the communication with the master control of AUG is realized by a programmable logic controller (PLC). The controlling of the multiple inverters is done by Linux-based industrial PCs. These PCs communicate with each other and to the PLC via a local network. The communication between the PLC and the Linux-based PCs is realized with an open-source-library. The integration of the new system into the master control system of AUG requires communication with many individual controls. The safety system of the torus hall and the control of the general energy supply have to be modified, too. The existing current converter, which is used to feed the DC-link, was typically applied for test bench coils. Therefore it was driven in current control mode only. For our application voltage control mode is required. Therefore the control concept of the converter had to be adjusted for the new application, while coil power supply still needs to be possible. With these boundary conditions, several feedback control concepts have been designed and tested. Also some modifications of the power stack were necessary to realize save discharge of the DC-link. All these steps required to make use of the converter for the DC-link supply and to integrate the system into AUG operation, including results from commissioning and initial experiments are presented in the paper. This project has received funding from the Euratom research and training programme 2014-2018.

Id 210

Abstract Final Nr. P3.078

Electrical and mechanical adaptation of commercially available power inverter modules for BUSSARD - the power supply of ASDEX Upgrade in vessel saddle coils

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Presently at ASDEX Upgrade 16 saddle-shaped pertubation coils - so called B-coils – are installed close to the plasma. Up to now they are supplied with DC current by two 4-quadrant current converters. A power patch panel allows connecting the coils or groups of several coils in series in either polarity. Due to the JxB forces and the mechanical design of the B-coils, a strict current limitation of 2.2 kA has to be considered. To supply the B-coils with AC current up to 500 Hz, 1.3 kA and arbitrary waveforms, each coil is powered by an individual inverter fed by a common DC link. To keep cost and development time low, commercially available power modules are used and existing current converters feed the DC links. Three power modules are mounted in one cubicle for realising a three level neutral point clamped (NPC) topology with lowest possible inductivity and making the most of the limited space available. In a first step four inverters are built to feed four coils each in series, each with reduced bandwidth, or four single coils with full bandwith. The paper presents the effort and steps required to adapt standard power blocks towards the needs of the ASDEX Upgrade power supply as well as the mechanical optimisations for good mountability, flexibility and scalability. Besides, solutions for mandatory personel safety and plant safety are presented. Coil damage by overcurrent or a high energy arcing in a vacuum insulated feedthrough would require a technically challenging repair with long downtime and the associated expense. This project has received funding from the Euratom research and training programme 2014-2018.

Id 208

Abstract Final Nr. P3.079

Requirements for qualification of manufacture of the ITER Central Solenoid and Correction Coils

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The Final Design of the Central Solenoid (CS) of the ITER Magnet system is now being completed and the manufacturing line of the modules under installation at the supplier's premises in the USA. The Final Design of the Correction Coils (CC) was completed in 2010 and installation of the manufacturing line at the supplier's premises was completed in 2013 in China. Manufacture of the first CS module and of the first CC are similarly planned to start in 2014. In order to meet specified magnet system performance, design requirements have been established by the ITER Organization. Qualification of the manufacturing procedures is thus a key point aiming at the demonstration of the achievability of these properties before starting effective manufacture of the first coils. The qualification phase of the manufacturing procedures includes manufacture of a set of relevant mock-ups using the actual manufacturing tools and the planned manufacturing procedures. These mock-ups are further submitted to a set of tests and qualification is declared successful when acceptance criteria are met. Depending on each item to qualify, mock-ups and trials are either defined according to standards or codes when they specifically address the qualification topic or according to specifically developed design. Acceptance criteria are in each case developed in such a way allowing on one a hand achievability of the required performance and on the other hand high enough margin with respect to property limits. The paper describes the topics to be qualified for starting coil manufacture, the requirements set for each topic, the mock-ups and trials requested and the acceptance criteria developed for CS and CC coils. First qualification results are reported. The main items concerned are conductor winding, He inlet manufacture, joint manufacture, insulation and impregnation and structure machining and welding.

Id 217

Abstract Final Nr. P3.080

Manufacturing design and development of the current feeders and coil terminal boxes for JT-60SA

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Current feeder and Coil Terminal Box (CTB) for the superconducting magnets for JT-60SA were designed. Copper busbar from power supply is connected to the High Temperature Superconductor Current Lead (HTS CL), which is installed on the vacuum vessel called CTB. The superconducting current feeder is connected to the cold end of HTS CL, and is led to main cryostat. The current feeder in CTB is connected to the feeder in the cryostat through electrical joint called mid joint. The in-cryostat feeder is connected to the terminal joint of magnet. Since maximum allowable magnetic field for HTS CL is 33 mT, HTS CLs are 12 m away from torus center. The maximum allowable horizontal force for HTS CL is 560 N. The thermal shrinkage of current feeder by cooling down was dealt with as follows. The octagonal ring with four supports from room temperature is adapted for the fixing of current feeders to reduce the horizontal force of HTS CL. Current feeder in CTB bent to crank shape was designed to reduce the stress of support. Trial manufacturing of crank shaped feeder was performed to confirm the manufacturability. The connection work of terminal joint and mid joint has to be performed at narrow space in cryostat. Because the insufficient soldering of joint was concerned because the direction of joint is vertical, the small tool which can connect soldering joint with vertical direction was developed. Though it is difficult to pressurize and heat up the insulation tape on current feeder for curing, insulation materials made by manufacturing condition showed sufficient shear stress. Since the all manufacturing process concerned was confirmed, the production of current feeder and CTB can be started.

Id 233

Abstract Final Nr. P3.081

In-vessel coils for magnetic error field correction in JT-60SA

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JT-60SA is designed and under construction as fully superconducting tokamak under a combined project of the ITER satellite tokamak program of EU-JA (Broader Approach Activities) and the Japanese national program. One of the main purposes of JT-60SA is the steady-state high-beta operation above the ideal no-wall beta limit. To achieve this, we have designed in-vessel coils, thus error filed correction coils (EFCCs) for a correction of magnetic error fields that affect plasma initiation and induce magnetic island locking. In order to design the number of coils and coil currents, we have probabilistically estimated the magnetic error fields from possible sources, thus manufacturing error and assembly misalignment of toroidal and poloidal field coils (TFCs/PFCs), the neutral beam (NB) field compensation systems and so on. The estimated error fields at the q=2 rational surface are about 5, 3 and 2 Gausses of m/n=1/1, 2/1 and 3/1 components, respectively. To have flexibility to correct magnetic field pattern including multi-components, three sets of saddle coils, toroidally 6 times poloidally 3 coils, are designed. Each coil consists of 35 turns of copper with a maximum current of 30kA turn for the upper and lower coils and 45kA turn for the middle ones. The EFCCs are also utilized to make resonant magnetic field perturbations (RMPs) for the magnetic stochastization for edge localized mode (ELM) control. We have also estimated the magnetic stochastization by the EFCCs by magnetic field line tracing with superposing the EFCC vacuum magnetic fields. It is found that the EFCC current higher than 10kA turn is enough for magnetic stochastization at the edge region under vacuum approximation. We will report the design of the EFCC in JT-60SA from an engineering and a physics points of view.

Id 55

Abstract Final Nr. P3.082

Accurate 3D modeling of Cable in Conduit Conductor type superconductors by X-ray microtomography

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Operation and data acquisition of an X-ray micro-tomograph developed at INFLPR are optimized to produce stacks of 2-D high-resolution tomographic sections of Cable in Conduit Conductor (CICC) type superconductors demanded in major fusion projects. High-resolution images for CCIC samples (486 NbTi&Cu strands of 0.81 mm diameter, jacketed in rectangular stainless steel pipes of 22x26 mm^2) are obtained by a combination of high energy/ high intensity and small focus spot X-ray source and high resolution /efficiency detector array. The stack of reconstructed slices is the input data for quantitative analysis consisting of accurate strands positioning, determination of the local and global void fraction and 3D strand trajectory assignment for relevant fragments of cable (~300 mm). Strand positioning algorithm is based on the application of Gabor Annular filtering followed by local maxima detection. The local void fraction is extensively mapped by employing local segmentation methods at a space resolution of about 50 sub-cells sized to be relevant to triplet of triplet twisting pattern. For the strand trajectory assignment we implemented two types of algorithms: i) a simple local algorithm trying to match the strands in adjacent slices. Even with a strand positioning efficiency over 95% such a simple algorithm cannot assign more than 50% of strands trajectories; ii) a global algorithm of the linear programming type which provides dramatically improved number of strand trajectories. For the benchmark CCIC samples 99% of the trajectories are correctly assigned. For production samples the efficiency of the algorithm is around 90%. Trajectory assignment of a high proportion of the strands is a crucial result for the derivation of statistical properties of the cable such as twisting pattern, cos(theta) or void fraction.

Id 531

Abstract Final Nr. P3.083

Development activities of the High Heat Flux Scraper Element

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The actively water-cooled divertor of the stellarator Wendelstein 7-X, at present under manufacture, will be installed in the machine before start of operation in 2019. Recent plasma simulations showed that, for certain plasma scenarios, the evolution of bootstrap current generates heat fluxes on the divertor element ends beyond their technological limits of 5 MW/m². To reduce the heat loads at these locations, a scraper element is being investigated to intercept some of the plasma fluxes before they reach the surface of the ten discrete divertor units. Each of the ten similar scraper elements must withstand a 550 kW steady-state heat load, with localized fluxes as high as 20 MW/m². Each element is actively water-cooled and has a surface area of 0.17 m² that is contoured in order to optimally intercept both upstream and downstream plasma fluxes. The main components of the scraper element are: 24 scraper fingers, 2 manifolds for inlet and outlet flows, a back plate with limited adjustment capabilities, and a support connection system to the plasma vessel. The scraper finger design is based on the monoblock technology: 13 monoblocks made of CFC NB31 are joined onto a CuCrZr cooling tube equipped with a copper twisted tape. Each finger is 247 mm long and 28 mm wide. The paper presents the current design of the scraper element and its development status aimed at validating the selected geometry and the manufacturing process with the relevant quality controls.

Id 119

Abstract Final Nr. P3.084

Conceptual design of the W7-X in vessel port protection for steady state operation

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The experiment WENDELSTEIN 7-X is a stellarator for steady-state operation with a pulse length of up to 30 min. The plasma vessel wall and ports require actively cooled protection due to the high thermal loads expected during plasma operation. IPP is presently working on the design and development of the port protection components that shall be installed when, after the first operation phase, the machine is refitted for steady-state operation. The work package has reached the conceptual stage. The functional requirements for the port protection liners are explained and the available options are shown. The conceptual design to meet the requirements is described, together with the technological solutions selected for the components. The presentation will conclude with a description of how the technological solutions will be implemented, including the use of full-scale technology demonstrators, leading to the final components.

Id 129

Abstract Final Nr. P3.085

Water-cooling system of the W7-X plasma facing components

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The water-cooling system of the plasma facing components of the Wendelstein 7-X stellarator was originally conceived for long pulse (up to 30 minutes) plasma operation based on an input plasma power of 10 MW. Over the last few years, the planning of the project has been revised and two intermediate machine operation phases have been introduced prior to the completion of the full long pulse capability. These are: OP 1.1, the first plasma operation phase with a plasma duration < 1s and 2MW input power and OP 1.2, the first divertor operation phase with a plasma duration of 5-10 s and up to 8MW input power. OP2, the steady state plasma operation phase with plasma duration of up to 30 minutes and 10 MW input power, will follow after the completion of these intermediate phases. The OP 2 piping system of the plasma facing components will be mounted from the start of operation (OP 1.1) but it has been adapted for the intermediate operational phases. In OP1.1 cooling of the glow discharge electrode housings is needed and in OP1.2 cooling of significant parts of the first wall is necessary to protect the components themselves, the plasma vessel and diagnostics from the loads of the heating systems. The use of the full piping system will be implemented in OP2. The paper presents the solutions selected to adapt the piping system to the different operation phases of W7-X and the resulting characteristics of the water cooling system as well.

Id 138

Abstract Final Nr. P3.086

Mechanical examination and analysis of W7-X divertor module sub-structures.

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For the long pulse operation phase, the W7-X stellarator is equipped with an actively water cooled high heat flux (HHF) divertor, consisting of parallel cooled target elements mounted in individual target modules. Due to the thermal deformation of these target elements during heat loading, the pipework that connects the target elements to the water supply manifold is subject to significant forces. Finite element calculations, for target modules TMh7-TMh9, show the superimposed forces of the whole pipework structure on to the manifold resulting in a torsional torque on the manifold support structure and weld. During manufacture, welding of the manifold to its support structure produces thermal induced distortion, resulting in difficulty in maintaining the accuracy of the manifolds. The welding between manifold and support structure was thus minimised in order to reduce this distortion. Finite element calculations showed that the nominal welds were acceptable; however, mechanical stress test on the manifolds mount point was carried out to prove the weld performance under the calculated loading conditions to ensure the safety of the component. For the remaining modules under design TMh1-TMh4 a parametric finite element calculation design study on the effect of the pipe length and routing on the stiffness helped to define minimum requirements for the design. The status of the manifolds for these modules will be shown. The manifolds are also mechanically connected to the port plug-in, therefore the impact of the thermal displacements on this pipework coming from plasma radiation affecting the target elements and from power loads coming from ECRH stray field radiation have been calculated. The paper discusses the results of the calculations and presents the outcomes of the stress test.

Id 152

Abstract Final Nr. P3.087

Experience gained from the 3D machining of the HHF divertor target elements

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The operation of the W7-X stellarator for pulse lengths up to 30 minutes with 10 MW of input power requires a full set of actively-cooled plasma facing components mostly covered with carbon-based materials. These components in W7-X have a plasma facing surface of about 265m². A vital part of the in-vessel components is the high heat flux (HHF) divertor which has a 19m² complex 3D formed surface consisting of 100 target modules assembled from 890 target elements. The divertor surface will be built up of individually 3D machined target elements with 89 individual element types. To date 300 of the 890 target elements have been 3D machined without geometrical problems. The qualification, development of acceptance criteria, testing and process of the 3D-machining of these complex elements will be explained and presented, describing the measures taken to minimize the risk of unacceptable damage during the manufacturing. After the 3D-machining, during the incoming inspection, copper infiltration from the CFC/CuCrZr interface to the plasma facing surface was detected in a small number of elements. The procedure for the further use of these target elements will be presented.

Id 166

Abstract Final Nr. P3.088

Results of high heat flux testing of W/CuCrZr multilayer composites with percolating microstructure for plasma-facing components

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Joining tungsten to copper is a major issue in the design of water-cooled divertor plasma facing components for future fusion reactors. The strong mismatch of the coefficients of thermal expansion of both materials generates high stresses during high heat flux (HHF) loading. A W/Cu functionally graded interlayer reduces the thermal stresses significantly. As a new solution, the development of infiltrated W/CuCrZr composites increases the strength at elevated temperatures compared to a pure Cu matrix and extends the operation temperature. A tri-layered W/CuCrZr joint (W70,W50,W30%vol) with mutually percolating microstructure of the two phases was developed as novel interface. The thickness and the amount of CuCrZr in the functionally graded material is a result of microstructure-based analysis of thermal- and mechanical behaviors [1]. A design optimization study was carried out by means of nonlinear finite element analysis. The optimal concentration profile of the interlayer across the thickness was identified. The optimization of the three layer system was performed in such a way that the resulting maximum equivalent stress intensity in the composite was minimized. This interlayer is manufactured by infiltration of CuCr1%Zr melt into the tungsten skeleton, subsequently annealed, quenched and aged to attain precipitation hardening [2]. The sintered skeleton with nearly 100% open porosity and varying W-composition was powder metallurgically manufactured. Three flat tile mock-ups with W/CuCrZr multilayers has been tested in HHF facility GLADIS. W-Tiles and multilayers were joined onto an actively water-cooled heat sink made of CuCrZr via electron beam welding. Cycling tests at 10 MW/m² and screening tests up to 20 MW/m² were successfully performed and confirmed the expected thermal performance of the compound. The thermal behaviour is in good agreement with the FEM predicted temperatures during the heat flux loading. Furthermore, microscopic investigation showed that the implementation of the novel functionally graded interlayer was successful. [1] A. Zivelonghi et al., Journal of Nuclear Materials 417 (2011) 536–539 [2] S.Nawka, et al., Proc. Powder Metallurgy World Congr.(2010) Vol.5 pp 383-90

Id 618

Abstract Final Nr. P3.089

Final tests and structural analysis of the new Solid Tungsten Divertor Tile for ASDEX Upgrade

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Tungsten as plasma-facing material for fusion devices is currently the most favourable candidate material. The eligibility of tungsten for needs in plasma fusion science has been intensively investigated at the fusion experiment ASDEX Upgrade (AUG) for over 14 years. Starting with the experimental campaign 2007 AUG was operated as a full tungsten experiment. The next step in the divertor improvement was the installation of a solid tungsten divertor, Div-III, to increase the thermal performance of the divertor tiles. The entire rebuilding with instalation of the solid divertor tile have been complited at the end of 2013. The themal performance of the solid tungsten tiles has been extensive tested in the high heat flux test facility GLADIS during the developing phase. To simulate the thermal loading due to high power plasma operation in AUG, cyclic loading tests have been performed up to 200 cycles. This corresponds to about 4 years of operation with about 50 high power discharges per campaign. The applied loading profiles are Gaussian with central heat flux of 10 - 30 MW/m2 and absorbed energy between 370 and 780 kJ, simulating the expected highest averaged power and energy loads in AUG. This loading results in maximum surface temperatures between 1500°C and up to 2800°C. The tests have been accompanied by intensive Finite Element Analysis. A multiple non-linear finite element model has been used. The elastic-plastic material response constitutive law was used for the tungsten tile. Contact surfaces were used for the geometrical separation of the model components. The elastic-plastic calculation was applied to analyse thermal stress and the observed elastic and plastic deformation during the heat loading. This paper discusses the main results of the GLADIS tests and their numerical simulations. Moreover, first results from the operation in the AUG experiment will be presented.

Id 591

Abstract Final Nr. P3.090

Plasma-wall interactions with nitrogen-seeding in all-metal fusion devices: formation of beryllium nitride and ammonia

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In all-metal fusion devices extrinsic radiating species are seeded into the divertor region to reduce the power load onto the divertor target plates. Nitrogen is routinely and efficiently used for this purpose in AUG and JET. The chemical reactivity of nitrogen can influence plasma-facing material properties, material erosion and hydrogen isotope retention. In this paper we expose several consequences of the use of nitrogen as a seeding species and investigate potential operational and safety issues. Upon implantation of nitrogen ions at keV energies into Be - the main chamber first wall material foreseen for ITER - a N-enriched surface layer evolves with a thickness limited to the implantation range, i.e. to a few nm [1]. Upon annealing of the implanted sample to 1000 K no diffusion of N is observed. These results alleviate concerns about modified properties of the bulk material. Be erosion has been measured in-situ with the quartz microbalance technique [2]. The resulting sputtering yields exclude a strong chemical enhancement at the applied N energies in the keV range. By interaction of nitrogen and hydrogen ions and/or radicals ammonia can be formed. This reaction is dominated by the catalytic activity of the metal surfaces of the vacuum vessel. In AUG the conversion of 5 % of the injected N atoms into ammonia has been observed in strongly seeded discharges [3]. Such a high production rate of tritiated ammonia would necessitate an additional separation step in the tritium plant of ITER. The database on AUG has now been extended to different discharge scenarios and is compared to first results on ammonia production obtained in JET which appears to be in the same order as in AUG. [1] M. Oberkofler and Ch. Linsmeier, Nucl. Fusion 50 (2010) 125001 [2] K. Dobes et al., proceedings ISI-21 (2013) vol.1, p.83, http://isi2013.spbstu.ru/eng/frst_en.html [3] V. Rohde et al., proceedings EPS-40 (2013) P2.123

Id 270

Abstract Final Nr. P3.091

Long term Project in ASDEX Upgrade: implementation of ferritic steel as in vessel wall

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A long term project recently started in ASDEX Upgrade (AUG) tokamak is the exploration of the compatibility of low activation ferritic steel with fusion devices. The topic is oriented towards the preparation of future experiments such as ITER with its test blanket modules and DEMO with its first wall designed with Eurofer. The goal of the project is to gather experience with ferromagnetic materials inside the vacuum vessel, dealing with magnetic perturbations, both in plasma and magnetic probes, and facing up the additional magnetic forces acting on the supporting structures. For the time being, the main AUG actor is the inner heat shield, but further development can be imagined in the future. The project is focused on the replacement of the graphite tiles supported by the heat shield with ferritic steel. The work activity has been split in 2 main steps: 1. Replacement of just two tile rows, symmetrically placed with respect to the plasma center position, in order to reduce the asymmetric field effects in plasma; 2. Extension of the metal wall replacing the whole graphite tiles, which may trigger a re-design of the heat shield support to stand the magnetic forces. The martensitic steel P92 was selected for these activities as an alternative to Eurofer, to facilitate the procurement and to cut expenses by a factor 5 for the raw material. To cope for the lack of data available for the P92, a magnetic characterization of the material has been carried out, confirming the suitability of the choice. The first phase has been already accomplished and presently 2 rows of bulk steel tiles are installed inside the AUG vacuum vessel. In preparation of this phase, an electromagnetic ANSYS 3D finite element model of one sector of AUG has been developed to pursue the above mentioned objectives (magnetic perturbations and forces). A corresponding structural model of the inner heat shield supporting structure was modelled to address the problem from a mechanical point of view. The experimental campaign has just started and, for the time being, has not suffered any particular problem related to the perturbation field induced by the steel tiles, as predicted by the calculation hereby reported.

Id 806

Abstract Final Nr. P3.092

Efficiency of water coolant for DEMO PFC

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An efficient removal of the high amount of energy at operating temperatures compatible with structural materials is an important issue for DEMO reactor and can be resolved by selection of a suitable coolant. Up to now, water-cooled divertor concepts have been developed for limited incident fluxes without taking into account transient power loadings. As a matter of fact DEMO concepts must be tested for much higher incident fluxes which are expected in H-mode operation with edge localized modes [1]. In this paper we analysed the efficiency of water as a coolant for the particular PFC tungsten monoblock shield with a cooling tube made from Cu alloy (Cu OFHC) as a laminate adjacent to W and a low activation martensitic steel (EUROFER) as inner tube contacting the coolant. EUROFER has been selected because it is the reference candidate structural material due to its expected capability of withstanding neutron damages higher than 70 dpa, corresponding to 3–4 years of DEMO divertor continuous operations. Thermal analysis is carried out by using the code MEMOS, which simulates W armour damage under the repetitive ELM heat loads. Heat transfer to the water coolant, including the regimes of single phase convective cooling and of sub-cooled boiling, is studied for various water conditions. The critical heat flux (CHF), increased by a swirl, is estimated using various correlations. We consider cooling conditions which allow one to keep relatively high material temperatures (in the range 300–600oC) thus excluding the EUROFER embrittlement and suffering of W from irradiation hardening. [1] Yu. Igitkhanov et al., submitted to the IEEE Transactions on Plasma Science, 2013

Id 128

Abstract Final Nr. P3.093

Mechanical testing of joints processed by electro-plating technology for brazing of divertor components

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Reliable and adapted joints are a challenge in divertor development independently of cooling medium and design. Tungsten will always be the armor material which has to be joined to heat sinks under functional and structural aspects considering the metallurgical interactions of alloys (tungsten, steel or copper) to be assembled and the filler materials. Application of conventional brazing tools and fillers showed lacks ranging from bad wetting of tungsten up to embrittlement of fillers under operation conditions. Thus, the development of electro-plating technology and deposition of reactive interlayers, e.g. Ni or Pd was initiated to overcome these restrictions Metallurgical testing of tungsten joints showed encouraging potential applying this technology. However, joints suitable for divertor application have to exhibit additional characteristics like aligned mechanical behavior. In this paper the joining of tungsten, Eurofer and stainless steel will be outlined applying electro-plating technology for deposition of interlayers and filler metal copper. The processed joints were analyzed in the conditions as brazed and aged for some 100 to 1000 h at a temperature of 700°C to obtain life time dependent values. The mechanical characterization was performed by applying shear testing to analyze the main type of loading condition under application. The testing was conducted at room temperature and at elevated temperatures up to 450°C. Shear strength was evaluated to be above 200 MPa and comparable with values known from conventional brazing. Cracks occurring in the Cu-Ni based filler showed ductile behavior at room temperature for all alloy combinations. The results confirmed the potential of the developed joining technology by electro-plating and will be discussed in detail. Looking forward to tungsten-copper joints as required in water cooled systems reactivity of the filler – interlayer system has also to be qualified under bonding / diffusion conditions beyond filler melting in future process development and qualification.

Id 354

Abstract Final Nr. P3.094

Basic considerations on the pump-down time in the dwell phase of a pulsed fusion DEMO

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Europe is currently working on the conceptual design of a demonstration power plant (DEMO) that shall be operated under quasi stationary conditions and shall deliver electric energy to the grid. In order to allow an efficient and economically attractive operation scenario, the length of the plasma pulses (burn phase) shall be maximized and the time in between the pulses (dwell time) must be reduced to an absolute minimum. The dwell time to provide the necessary start vacuum (of the order of $5 \cdot 10^{-4}$ Pa) for the next plasma discharge is requested to be not longer than the time to load the central solenoid, which is expected to be of the order of 30 min for DEMO. The evacuation of the vacuum vessel is counteracted by outgassing of helium and hydrogen isotopes that were absorbed on or solved in the plasma facing materials in the previous shots. For the material, we assume tungsten in various forms. The gas release is a very complex process and is given by a number of different, partly competing mass transfer processes based on solubility, diffusion, recombination, desorption and trapping, which depend strongly on temperature. The paper takes four basic approaches to this problem, however, all of them have their limitations. As first step, the fundamental aspects are introduced. It will be shown that any quantitative prediction based on theoretical estimations suffers from a large uncertainty. Secondly, an empirical formulation of the outgassing term will be used to solve the pump-down equation and to discuss physics boundary cases. Thirdly, supporting outgassing experiments are performed with different tungsten samples. And lastly, comparisons are made with existing pump-down data from ASDEX Upgrade. In conclusion, a window of expected dwell times is figured out, which shows that the 30 min requirement may be very difficult to reach.

Id 355

Abstract Final Nr. P3.095

Magnetohydrodynamic flows in model porous structures

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Liquid metals have been identified as potential solution for many engineering problems associated with the use of solid materials as plasma facing components (PFCs) in a fusion reactor. A promising option for applications to divertors seems to be the use of capillary porous structures saturated with a liquid metal, on which a stable liquid surface could be obtained by capillary forces. The aim of the present study is providing a first description of liquid metal flows in a capillary porous system (CPS) when an external uniform magnetic field is imposed, in order to judge about the feasibility, from the magnetohydrodynamic (MHD) point of view, of the CPS technology for steady-state divertor applications. The final objective is estimating a permeability tensor that relates the average fluid velocity to the mean pressure gradient as required for a macroscopic description of the problem. The influence of the strength and orientation of the magnetic field on pressure losses in the porous structure is assessed for a generic model geometry. The permeability is determined through numerical simulations of liquid metal flows in a microscopic representative fluid volume. The numerical results show that, independently of the orientation of the magnetic field, the permeability strongly reduces in a MHD flow compared to the one in the corresponding hydrodynamic Darcy regime. Main parameters that influence MHD flows in the CPS are identified.

Id 764

Abstract Final Nr. P3.096

SIRHEX - a new experimental facility for high heat flux testing of plasma facing components

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In the past years the Karlsruhe Institute of Technology (KIT) has developed manufacturing technologies for Test Blanket Module (TBM) components especially for the so called First Wall (FW). Facing the fusion plasma directly, the FW is subject to cyclic loads of high surface heat flux. In order to investigate its thermo-mechanical behaviour under these conditions and to qualify the manufacturing procedure it is foreseen to test a reduced scale FW mock-up in the Helium Loop Karlsruhe (HELOKA) facility at KIT. To reproduce the high heat loads, a surface heater with fast response time and capable of generating a homogeneous surface heat flux up to 500 kW/m² is required. Following an investigation of different heating methods it has been chosen to use an infrared radiation heater consisting of a tungsten filament inside a quartz glass tube with a reflective coating on one side of the tube. As the heated surface area is fairly large (0.34 m²) several such tubes have to be employed to achieve a homogeneous heat distribution. In order to qualify the heater and to find the best configuration of the tubes, a new small scale experimental facility has been built at KIT, called SIRHEX ("Surface Infrared Radiation Heating Experiment"). In SIRHEX an instrumented water cooled target is heated using three heater tubes. The distance between the tubes as well as the distance of the heaters to the target can be varied in order to determine the best tubes configurations. In parallel with the experiment an ANSYS model of the heaters has been developed. It will be qualified and optimized based on the results of the SIRHEX experiments and used to find the optimal configuration of heater tubes for FW experiments. This paper describes the SIRHEX facility and the experimental set-up for the heater tests. The results of the first series of tests will be presented and the impact of the heater performance on the design of the FW test rig will be discussed.

Id 867

Abstract Final Nr. P3.097

Modeling and Optimization of graded tungsten/EUROFER97 coating system for First Wall components

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Reduced activation Ferritic/Martensitic steels (RAFMs) (e.g. EUROFER97) are to be used as structural material for the First Wall (FW) of future fusion power plants. The interaction (especially physical sputtering) between plasma and FW will limit lifetime under normal operation. Tungsten coating is chosen to protect the FW due to its low sputtering yield and low activation. However, the mismatching of thermo-physical properties between tungsten and EUROFER97 can lead to large residual thermal stresses. In this work, erosion protective tungsten coating with tungsten/EUROFER97 functional graded (FG) interlayer on EUROFER97 substrate will be developed and optimized. The coating as well as the FG interlayer will be produced by Vacuum Plasma Spraying (VPS) with parameters optimized by modelling and evaluated by means of microstructural and micromechanical investigations. For determining optimal parameters for tungsten coating and FG interlayer the fabrication phase and operation phase are simulated with the Finite Element (FE) Method ABAQUS. Thereby the influences of layers thicknesses and graded function on relieving residual stress and reducing inelastic strains are investigated considering plasticity and creep of the materials. Based on the FE results creep-fatigue assessment of the coating system is performed demonstrating the gain in lifetime to be expected when using a FG interlayer and investigating its dependence on the thickness of the FG interlayer. In addition to the modelling results the planed experiment s for fabricating the coating systems will be reported and discussed.

Id 837

Abstract Final Nr. P3.098

Results of first wall manufacturing at KIT

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The manufacturing of one of the most challenging components of a reactor is the so called First Wall (FW). The realization of a fabrication procedure including the qualification routines according to applicable codes and standards is the goal of an activity supported by national funding (BMBF) in the KIT. The FW faces the breeding blanket against the plasma as heat exchanger, plasma impact protection and structural element. One fabrication option for the FW investigated in the KIT is to assemble the semi- finished First Wall by a diffusion bonding process from two plane half plates with a symmetric half deep in milled coolant channel pattern. Then a forming process will be applied to realize the two 90° bends to create the corners of the component. After investigation in medium scale now relevant scale components have been fabricated in industry. The recent experimental series included 5 differently sized FW mock ups in dimensions from 400 mm x 710 mm, up to about 580 mm x 2900 mm which corresponds to a full scale component for a ITER TBM application. The paper reports the present results of the applied non-destructive and destructive qualification routines according to the codes and standards.

Id 848

Abstract Final Nr. P3.099

Comparison of EM loads acting on DEMO blankets under possible design-driven electrical connections

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Off-normal operations in Tokamak reactors result in the induction of eddy currents in the electrically conductive components that, coupled with the large magnetic field, impose strong electromagnetic forces (Lorentz's forces) to fusion reactor components. These forces are strongly dependent on the event considered as well as on the electrical connections between the different conductive components of the reactor. Since they constitute a severe issue for the mechanical structure, a careful analysis of the conditions that can affect their value has to be performed in order to consolidate the present reactor knowledge and technology into a reference design. The study performed in this work is aimed to assess the EM loads acting on a single blanket module as well asand on the complete blanket segment. Different plasma scenarios and electrical contacts between modules, manifold and vacuum vessel are taken into account. In particular, the electrical assumptions used in this work are based on possible design solutions proposed for the module attachment and segment remote handling. To this purpose, three different EM models have been implemented upgrading the model used in the past previous works campaigns on the basis of the DEMO reactor configuration elaborated by EFDA in 2013. EM analyses have been performed using the commercial ANSYS© code and considering both the HCPB and HCLL concepts for the breeding zone of the modules. The analyses take into account for the ferromagnetic properties of the EUROFER material in the blankets composition and, in comparison with the previous works, Maxwell forces have been considered in the results.

Id 999

Abstract Final Nr. P3.100

Fabrication and integrity test of the HIP joined W/FMS mock-ups for developing the Korean DEMO PFC

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Tungsten (W) as armor and Ferritic-Martensitic (FM) steel as a structural material are the major candidates of Plasma Facing Components (PFCs) such as the blanket First Wall (FW) and divertor in a fusion reactor. In the present study, the W/FMS joining method was investigated to apply to the PFC for a fusion reactor. Three W/FMS mock-ups were fabricated with Hot Isostatic Pressing (HIP, 950 oC, 100 MPa, 1.5 hrs) with a following post-HIP heat treatment (tempering, 750 oC, 70 MPa, 2hrs), methods that were based on the ITER BFW and Test Blanket Module (TBM) development project from 2004 to the present. By an ultrasonic test to the joint with a 10 MHz frequency flat type probe, it was found that there was no defect in the fabricated mock-ups. To confirm the joint integrity, a high heat flux test was performed up to the thermal life-time of the mockup with the proper test conditions (2500 cycles under 1.0 MW/m2 heat flux) using the Korea Heat Load Test facility with an Electron Beam (KoHLT-EB), and its water coolant system at KAERI. The test conditions were determined through a preliminary analysis with conventional codes such as ANSYS-CFX for thermal-hydraulic.

Id 463

Abstract Final Nr. P3.101

Heat Load Test for the Plasma Facing Components by using KoHLT-EB facility

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Plasma facing components are the dominant topics in the development of fusion reactors. The main components of the tokamak PFCs are the blanket first wall, divertor, and various diagnostics ports, which include the armour materials, the heat sink with the cooling mechanism, and the diagnostics devices for the temperature measurement. The Korea Heat Load Test facility, KoHLT-EB (Electron Beam) has been operating for the plasma facing components to develop fusion engineering in Korea. This electron beam facility was constructed using an 300 kW electron gun, and the maximum target dimension is 70 cm × 50 cm in a cylindrical vacuum chamber of about 140 cm in diameter and 250 cm in length. The performance tests were carried out for the calorimetric calibrations with Cu dummy mockup and for the heat load test of tungsten first wall mockups. Also the thermo-hydraulic tests were performed to evaluate the high temperature gas-cooling devices. For the simulation of the heat load test of each mockup, preliminary thermal-hydraulic analyses with ANSYS-CFX were performed. For the development of the plasma facing components, test mockups were fabricated and tested in the high heat flux test facility and non-destructive method. These manufacturing technologies and test performances will be used for the development of a fusion reactor.

Id 491

Abstract Final Nr. P3.102

Evaluation of thermal structural behavior of divertor under ELM

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The time scale of ELM has been measured to be several hundred μs in a number of devices and configurations. Large heat load would be applied to the target surface of divertor due to the characteristics short time scale. In this study, thermal structural behavior of divertor under ELM was evaluated by using finite element method. Tungsten monoblock was used as representative divertor model. Average of total heat flux, peak energy flux of ELM, and pulse width of ELM was considered as parameters in analysis. Single pulse of ELM was additionally applied after 10MW/m2 of average of total heat flux was applied to target surface. Tungsten melting will not occur even though peak heat flux of ELM increased until 0.83MJ/m2. However, thermal stress due to the local heating at tungsten surface exceeded tensile strength of tungsten even though applied peak energy flux was below 0.83MJ/m2. It would cause damage of divertor, such as plastic deformation, or crack. The result of the analysis suggests that the large heat load would lead to not only recrystallization, melting of tungsten, but also thermal stress. Dynamic heat load experiment that simulates the ELM was conducted for the comparison with the results of analysis. YAG laser was used to heat resources, and surface temperature and average temperature of specimen were measured by using pyrometer and thermocouples, respectively. When approximately 0.5MJ/m2 of peak energy flux was applied, recrystallization and cracking occurred at the tungsten surface. Melting did not occur within allowed heat load of divertor in ITER, 0.5MJ/m2. However, the results suggest that the divertor is anticipated to be damaged by thermal stress, and it would be possible to reduce life time by thermal fatigue. Therefore, recrystallization, thermal stress should be also considered for steady state plasma operation.

Id 139

Abstract Final Nr. P3.103

Hydrogen gas driven permeation through tungsten deposition layer formed by hydrogen plasma sputtering

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Evaluations of tritium inventory in the plasma confinement vessel and tritium permeation rate to a coolant are important issues from a viewpoint of safety. Some reports have shown that tungsten deposition layers can trap a certain amount of hydrogen isotopes in the process of growing [1,2]. However, basic behaviors of hydrogen isotopes in the tungsten deposition layers have not been understood quantitatively so far. In this work, tungsten deposition layers, which were formed by hydrogen RF plasma sputtering, were exposed to hydrogen gas and permeation behavior was observed. Two samples of tungsten deposition layers were formed on circular substrates of nickel. The thickness of deposition layers were 250 and 710 nm, respectively. The secondary side connecting to a pressure gauge was closed in a vacuum before a sample gas containing hydrogen (1,20,100%) was supplied to the primary side. Hydrogen permeation flux was obtained from the pressure rise in the secondary side at the temperature range from 148 to 520 o C. Hydrogen permeation flux in the sample was 3 orders smaller than that of nickel. One dimensional-diffusion in the deposition layer was numerically calculated and solubility and diffusivity were obtained by fitting calculated curves to experimental ones. The obtained solubility was several orders of magnitude higher than that in tungsten bulk and the obtained diffusivity was a few orders of magnitude smaller than that in tungsten bulk. The permeability, which is obtained by the production of solubility and diffusivity, was larger than that in tungsten bulk. From these results it can be said that the permeation flux of hydrogen through the tungsten wall in steady state does not decrease by the formation of the deposition layer although tritium inventory increases. [1] K.Katayama, et al., Fusion Sci. Technol. 54 (2008) 549. [2] G.De Temmerman and R.P. Doener, J.Nucl.Mater. 389 (2009) 479.

Id 487

Abstract Final Nr. P3.104

Surface modification of tungsten materials by repeated high heat loading

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Tungsten (W) is the primary candidate for use as plasma facing materials/components (PFM/PFC). PFM/PFC will be subjected to heavy thermal loads in the steady state or transient mode combined with high energy neutron irradiation that will cause serious material degradation. In the present work, repeated pulse high heat loading experiments have been performed in order to investigate the damage characteristics of tungsten materials caused by the repeated heat loading during ELM. ITER grade W, W-1.0wt%La2O3 and K(0.003wt%) doped W were machined to the dimensions of 10 mm x 10 mm x 1mm, followed by mechanical and electro polishing. All of the polished specimens were placed on a water-cooled Cu block and subjected to high heat load experiments by an electron beam irradiation test simulator. The experiments were conducted at the irradiation conditions; repeated irradiations of 2 second-irradiation and 7.5 second-rest with one cycle of 9.5 seconds for totally 200 times. The surface temperature of the samples changes from below 450oC to 1300oC by 2 second-irradiation. Before and after the irradiation, the specimen surfaces were examined by SEM. In addition, quantitative analyses about temperature profiles and elastic-plastic thermal stress have been carried out using FEA In the case of ITER grade W, the repeated irradiations of 20 times caused surface roughening in the intragranular. The surface roughening is due to plastic deformation caused by the thermal stresses due to temperature difference. The subsequent repeated irradiations of totally 200 times caused significant surface roughening, cracking in the intragranular and grain boundaries and surface exfoliation. This suggests that the repeated heat loading during ELM causes this kind of surface modification. Therefore, estimation of influence of the surface modification on erosion, exfoliation, lifetime, thermal property and hydrogen retention will be required. In addition, the profiles of thermal stress during and after the electron beam irradiation were evaluated by the comparison with the experimental and the results of the FEA analyses.

Id 1007

Abstract Final Nr. P3.105

Progress in the ITER TBM port plug design

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The validation and test of the design concepts of tritium breeding blankets, which are relevant for a future commercial reactor, is one of the goals of the ITER machine. To accomplish these objectives, mock-ups of breeding blankets, called Test Blanket Modules (TBMs), are tested in three ITER equatorial ports. Each TBM and the associated shield form a TBM-set that is mechanically attached to a steel frame. A frame and two TBM-sets form a TBM port plug (TBM PP). The ITER Organization is responsible for the design and manufacture of TBM frames and the dummy TBMs (which are replaceable with TBM-sets). The Conceptual Design Review (CDR) of TBM PP with two dummy TBMs was performed in the middle of 2013 to assess systems requirements, design analysis, interface requirements and manufacturing aspects. This paper describes the progress of the design of the TBM PP achieved after the CDR, as follows: (1) Improvement of attachment between the frame and dummy TBMs (applicable to TBM-sets), (2) Implementation of coverplates to avoid contamination during maintenance operation, (3) Reduction of clearance between the frame and dummy TBMs (applicable also to TBM-sets), (4) Modification of cooling layout and introduction of castellation design on the plasma facing area to minimize thermal stress, (5) Assessment of the cooling scheme to allow draining and drying efficiently, (6) Development of PP remote handling maintenance and assembly sequence including the operations to be performed in the hot cell, (7) Status of R&D to allow metallic gasket sealing in Dummy TBM (or TBM sets). The achieved design improvements result in a PP design that avoid all the concerns expressed during the CDR.

Id 144

Abstract Final Nr. P3.106

Novell manufacturing method by using stainless steel pipes expanded into aluminium profiles for the ITER neutral beam cryopumps

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The Heating Neutral Beam injectors require custom designed cryopumps which have pumping speeds of 4500 m3/s for H2 and 3600 m3/s for D2 operation. Such highly efficient cryopumps have been already developed in the past and optimum geometries have been developed to fulfill the pumping requirements given by the environment in the neutral beam injectors. To achieve the required pumping speeds for the ITER neutral beam injectors two cryopumps with an overall pumping inlet area of 38 m2 are installed in each injector. The cryopumps have a flat geometry and each of the two pumps has a length of 8 m, a height of 2.8 m and a depth of 0.45 m. Internally each cryopump contains 32 similar pumping sections made of three cryopanels for gas accumulation and four main thermal radiation shields to protect the cryopanels from high heat loads by thermal radiation. All these components need to be cooled and heated by a forced flow of pressurized helium and have extended fins to protect the cryopanels or to pump the gases. For one neutral beam injector several hundred of these components are required hence a reliable and efficient fabrication process has been developed. The use of stainless steel pipes expanded into aluminum extrusion profiles is a novel solution to combine standard stainless steel welding procedures for the piping with extended aluminum structures making advantage of the high thermal conductivity of aluminum. In this paper, the novel engineering and manufacturing solution for the thermal radiation shields and the cryopanels of the ITER neutral beam cryopumps is described. The investigations on the material stresses for a temperature range between 4 K and 400 K are outlined and it is shown how this has been proven by experiments during the manufacturing tests.

Id 151

Abstract Final Nr. P3.107

Distribution of the In-Vessel Diagnostics in ITER Tokamak

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The ITER Tokamak is divided into 9 similar sectors where all needed diagnostics are distributed to help understand burning plasma physics and assist in machine operation. The distribution requirements of the in-vessel diagnostics have been taken into account in the integration design. This paper will explain the distribution constrains for the highly complex in-vessel environment, different distribution criteria for each diagnostic, and the independent requirements and exclusions among them. In addition to the requirements for diagnostics themselves, there are also space limitations for the blanket shield blocks due to physical clashes or minimal clearance between systems. Besides, there are issues with the assembly sequences and gamma heating considerations that impose several additional restrictions in their allocations. Another particular restriction is that there are no zones in the First Wall and Shield Block to remove material without changing the blanket cooling channels or maintain the survival the blanket module. The in-vessel diagnostics with special constrains are: Magnetic pick-up coils • Tangential coils. Distributed behind all ITER blanket modules (BM's) with toroidal symmetry. • Normal coils. Distributed behind BM's on top of the ITER tokamak with toroidal symmetry • High-frequency magnetic sensors: - Toroidal array. Distributed toroidally around the tokamak torus at the same height (poloidal angel) - Poloidal arrays Distributed poloidally. - Port Array Sensors: On equatorial ports 1, 6 and 9. Neutron Activation System. The Irradiation ends of the system are in sectors 6 and 9. Pellet Injection System. The tube ends of the system are in sector 2, 5 and 8. Bolometers. There are 22 bolometers shared out for the Poloidal and Toroidal allocations. In this paper, the current solution for the In-Vessel Diagnostic Distribution will be presented. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

Id 237

Abstract Final Nr. P3.108

ITER Lip Seal Welding and Cutting Developments

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The large port plug assemblies form the main interface connection of the primary pumping, cooling, diagnostics, and fuelling systems to the vacuum vessel. When removed they also provide the maintenance access ports into the vessel. They are mechanically attached to the vessel with a combination of bolted flanges, and sealed with edge welded lip seals. For systems maintenance and machine access it is anticipated that their periodic removal will be required. The welded lip seals form part of the torus primary vacuum boundary, and are classified as Protection Important Component. It is therefore of primary importance to demonstrate the possibility to cut, weld and maintain these lip seals, in the port cell environment. The port plugs design must enable their removal and replacement for vessel and systems maintenance for up to ten times during the life time operation of the ITER machine. Therefore proven, remote reliable cutting and re-welding of the lip seals is essential, as these operations need to be performed in the port cells in a nuclear environment, where human presence will be restricted. Moreover, the combination of size of the components to be welded (~10m long vacuum compatible thin welds) and the congested environment close to the core of the machine constraint the type and size of tools to be used. This paper describes the lip seal cutting and welding development program performed at the VTT Technical Research Centre, Finland. Potential cutting and welding techniques are analyzed and compared. The development of the milling, TIG and laser welding techniques on samples are presented. Effects of lip seal misalignments and optimization of the 2 welding processes are discussed. Finally, the manufacturing and test of the two 1.2mx1m representative mock-ups are presented. The set-up and use of a robotic arm for the mock-up cutting and welding operations are also described.

Id 291

Abstract Final Nr. P3.109

ITER design features serving for suppression of eddy currents and electromagnetic loads

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Tokamak ITER will operate with high magnetic fields, in a scale of 10T. At the time of plasma disruption magnetic field variation rate reaches approximately 100T/s. Such transient fields induce intensive currents and electromagnetic (EM) loads in all conductive structures. Induced currents are split into two groups: eddy currents closed completely in conductive structures, and halo currents closed partly through plasma periphery and partly through structures. This article describes design features serving for suppression of eddy related EM loads in ITER. If the task is to suppress eddy related loads in massive parts, for example in blanket modules (BM), it is usually addressed with well-known design means such as re-shaping, slits and electro-insulating breaks in supports. Advanced methods employ unusual shapes resembling “wings” or “stars”. They are described in this paper. Different design means were used for suppression of net EM loads in electrical straps (ES): elastic conductive bridges crossing movable gaps between BM and vacuum vessel. Tight predefined space limits the range of possible options and the only acceptable solution was found with a principle of force-free and moment-free current paths. It is explained in this paper and may be utilised also for ES of port plugs and for elastic compensators in various bus-bars. Another solution was found for blanket manifolds. They include pipe bundles supported at the vacuum vessel and branch pipes welded to BMs. If such structure is fully metallic, it causes very high EM loads which it is unable to withstand. The only workable solution was found to insulate electrically each pipe from supports and other pipes. All-except-one legs of each support are also insulated. This solution suppressed EM loads approximately in one order of magnitude and allowed to develop a workable manifold system. Similar solutions may be used for other pipes running inside the tokamak.

Id 315

Abstract Final Nr. P3.110

Multi-Purpose Deployer for ITER In-Vessel Maintenance

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The Multi-Purpose Deployer (MPD) is a general purpose in-vessel remote handling (RH) system in the ITER RH System. The MPD provides the means for deployment and handling of in-vessel tools or components inside the Vacuum Vessel (VV) for dust and tritium inventory control, in-service inspection, leak localization, and in-vessel diagnostics. It also supports the operation of blanket first wall maintenance and duct liner module maintenance operations. This paper describes the design status of the MPD. The MPD is a cask based system, i.e. it stays in the hot cell building during the machine operation, and is deployed to the VV using the Cask System for the in-vessel operations. The main part of the MPD is the Articulated Transporter which provides transportation and positioning of the in-vessel tools or components. The Articulated Transporter has ten degrees of freedom with a high payload capacity up to two tons. The Transporter can position the tools on all the surfaces of the VV by switching among the four equatorial RH ports. Additionally it can use two non-RH equatorial ports to transfer port-sized tools or components. The tools or components are deployed using the Service Cask equipped with a Deployment Trolley. The Deployment Trolley can accommodate various Storage Boxes which are dedicated for end-effectors, tools, and in-vessel components. Among the end-effectors, the most versatile device is the dual arm Manipulator, which provides dexterous handling of the tools and components. It is deployed at the standard end-effector interface on the most distal joint of the Articulated Transporter, and can be positioned in any location within the VV. It has a force reflecting capability, an on-board hoist system, an articulated viewing system, and other static viewing systems mounted on its body or arms. The current conceptual design of the MPD demonstrates its feasibility to satisfy the baseline in-vessel task requirements.

Id 459

Abstract Final Nr. P3.111

First boundary electrical feedthroughs for the Heating Neutral Beams injectors of ITER

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The first boundary feedthroughs of the ITER Heating Neutral Beam injectors are key components of the neutral beam system. They allow the penetration of signals and power from/to the beam line components, through the confinement boundary of ITER. This article will present the status of their design as well as the safety requirements to be met. The first part of this article is dedicated to the presentation of the two types of feedthroughs (instrumentation and high voltage). The main design requirements are given before the presentation of the status of these components, which are at final design level. The cabling strategy used for the instrumentation allows flexibility to recover the signals from the beam line components in case of failure of the in-vessel cabling or remote handling connector. The design of both types of feedthroughs is based on commercial products arranged in a specific configuration to create a double containment and to limit common failure modes. These feedthroughs are compatible with the remote handling maintenance. This requirement comes from the activation level expected on these components. The second part presents the safety requirements for these components. As they use ceramic and brazed joints, which are not covered by mechanical standards, the demonstration of their robustness must be provided. This will be done through a control and testing plan which is under study. Beside mechanical analysis, the manufacturing and testing of these components will be carefully controlled. This quality control strategy will be based on existing standards adapted for these specific components. It will include the monitoring of the raw materials used, the control of each step of the manufacturing and the testing. The testing will include outgassing, pressure, leak and vibration tests to replicate seismic events. The present design satisfies all the requirements. It presents important margins to the allowable stresses, which would allow a successful qualification of these components.

Id 543

Abstract Final Nr. P3.112

Structural Damages Prevention of the ITER Vacuum Vessel and Ports by elasto-plastic analysis with regards to RCC-MR

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Several types of damages have to be prevented in order to guarantee the structural integrity of a structure with regards to RCC-MR; - the P-type damages which can result from the application to a structure of a steadily and regularly increasing loading or a constant loading. They are commonly associated by immediate or time dependant effects to the excessive deformation and structural instabilities, - the S-type damages during operational loading conditions which can only result from repeated application of loadings associated to the progressive deformations (the permanent overall deformation continues to increase as every loading cycle induces additional deformation and the structure gradually changed from its original shape) and fatigue (by progressive cracking). Some other phenomena exist such as the buckling or the fast fracture which are not strictly speaking a type of damage. For this reason these last one are not a part of paper's scope. Following RCC-MR, the S-type damages prevention has to be started only when the structural integrity is guaranteed against P-type damages. The verification of the last one on the ITER Vacuum Vessel & Ports has been performed by limit analysis with elasto-(perfectly)plastic material behaviour. This non-linear approach was employed to assess more realistic structural margins and thus to provide considerable margins as compensatory measures for the lack of In Service Inspections of a Nuclear Pressure Equipment, NPE level 2 cat IV associated to a Protection Important Component, PIC (including the Safety Class I). It is usual to employ non-linear analysis when the 'classical' elastic analysis reaches its limit of linear application. By the same way, some elasto-plastic analyses have been performed considering several cyclic loadings to evaluate also more realistic structural margins of the against S-type damages. The main results of this P-type and S-type damages prevention are proposed in this paper.

Id 604

Abstract Final Nr. P3.113

Diagnostic integration issues in the ITER Blanket First Wall

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ITER will have about 50 diagnostic systems for machine protection, and plasma control, as well as understanding the physics of burning plasma. Their implementation in the ITER device is challenging, particularly for the in-vessel diagnostics components, located in the region between the Vacuum Vessel and Blanket First Wall (FW) contours, where space is constrained by the high number of systems. This paper describes the current status of design integration efforts to accommodate diagnostics components in the ITER First Wall. These approaches are the basis for detailed optimization and improvement of existing concept design interfaces between systems. The aim of this study is to identify the critical issues related to the integration of FW diagnostic components that could affect the ITER FW performance or maintenance and address an engineering solution to successfully satisfy the requirements of the two parties It is possible to classify the ITER FW diagnostics in two different groups: - Optical diagnostics that required a plasma facing element often a mirror or beam dump/black body. This group is composed by the following diagnostic: Poloidal Polarimeter (mirror), Visible Infrared TV cameras (black body for calibration), H-Alpha (beam dump), Thomson Scattering (beam dump), - Dust and Tritium monitors which require a plasma facing removable sample. The main integration issue is accommodating these into the FW central slot, faced with the challenges of: - High nuclear heating, surface heat and electromagnetic radiation fluxes. - Space for integration is limited by compacted geometries of a challenging FW design. -Components difficult to remotely maintain and/or replace. Work is on-going, and in many cases, a mutually acceptable solution has been found. The current status will be presented. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

Id 590

Abstract Final Nr. P3.114

The choice of Dynamic Amplification Factors DAF's for the ITER Generic Port Plugs during disruptions

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The purpose of this paper is present an overview of the methodology followed to calculate the Dynamic Amplification Factors (DAFs) applied to the electromagnetic loads acting in the ITER Generic Port Plugs. The port plugs are safety important components, SIC-1, defined as "those required to bring and to maintain ITER in a safe state". The different single loads are classified depending of its probability of occurrence and, based on it, some load cases are built searching for an enveloping map of the possible damages. The method used to combine an EM transient event with another load is based in the application of this dynamic EM load as static. An initial transient dynamic analysis was performed, in the most loaded electromagnetic event, to determine the dynamic response of the Port Plug. In the same way, have been solved all the time steps of the dynamic event as static loads, it means that the inertial effect has been neglected. The dynamic response of each time-step has been compared with the same static step; for this purpose some control points were positioned along the structure. The key of the calculation is to understand how the port plug is deformed in order to obtain the Dynamic amplification factors that permit amplify the loads in a realistic way. A complementary modal analysis was performed to obtain the fundamental frequencies and vibration modes of the Generic Port Plug for the characterization of the damping of the structure. As conclusion, a conservative DAF of 1.23 was taken.

Id 644

Abstract Final Nr. P3.115

Starting manufacturing phase of the ITER upper ports

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The ITER Vacuum Vessel (VV) is an all-welded torus-shaped double-shell structure with stiffening ribs between the shells. The main function of the vessel and ports is to provide the high-vacuum and primary safety containment of the ITER machine, it also supports in-vessel components, such as the blanket modules and the divertor cassettes. To provide access inside the vessel for various needs, the VV features upper, equatorial and lower ports that are often occupied by the port plugs. The ports are also all-welded structures but include the double-wall and single-wall parts. As the ITER construction phase started, the procurement of the VV ports was launched. However, design of some remaining interfaces was still in progress: in particular of the Sealing Flange package, which includes the high-vacuum seals and the plug fasteners. Design of these components has intensely progressed towards finalization now. To study in detail the tightening process of the high-strength fasteners and manufacturing of the sealing components, an R&D programme has been launched. Following the Procurement Arrangements signed between the ITER Organization and Domestic Agencies (DAs), the VV Upper Ports are being procured by the Russian Federation DA. The RF DA has contracted the Efremov Institute (St. Petersburg, Russia) as the main supplier. In 2012 a world-wide call for tender for the supply of the Upper Port components was launched and the company MAN Diesel and Turbo (MDT, Deggendorf, Germany) was awarded the contract. All main port components are within the scope of this contract while the Efremov Institute remains responsible for the whole supply and performing various activities in support of the port manufacture (including material supply, engineering analyses, processing of the manufacturing models and drawings, etc.). The Upper Port manufacturing design, the results of related structural analyses and studies, and the fabrication progress are described in this paper.

Id 640

Abstract Final Nr. P3.116

Integration of remote refurbishment performed on ITER components

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Internal components in ITER are maintained and recovered from the Tokamak, by remote handling equipment. These components include port plugs, cryopumps, divertor cassettes, blanket modules, etc. They are brought to the Refurbishment Area of the ITER Hot Cell Building for cleaning and maintenance, using remote handling techniques. The Refurbishment Area is a substantial red zone room, with a 60 ton crane, independent mast, trolleys, cleaning cells, stainless steel liner, separate component storage room and a centralized detritiation system. The ITER Refurbishment Area is unique in the world, in that large quantities of both activated dust and tritium are present, carried on the components retrieved from the Tokamak. The low throughput of the ventilation requires that the large doors on the Red Zone are leak-tight, at a level comparable to a glove box. The integration covers a number of workstations to perform remote operations. In the cleaning cells, the components are remotely vacuumed to remove mobile dust, following which, the component is moved into the main portion of the Refurbishment Area. In the case of port plugs, a dedicated tilting station rotates the horizontally-oriented port plug into a vertical orientation, required for removing and installing modules. The vertical clearance required for maintaining the port plugs is substantial, particularly for the Upper Port Plugs. For this reason, the vertical clearance of the room is enhanced by a pit, into which port plugs are placed for maintenance and also storage. The Remote Handling equipment in the room (including the master-slave manipulators, cranes and trolleys) are maintained themselves in adjoining maintenance rooms, after passing through decontamination rooms. This paper describes the integration of the Refurbishment Area, explaining the function, the safety improvements and the trade-offs made.

Id 651

Abstract Final Nr. P3.117

Engineering constraints due to the ESP/ESPN regulation applied to the Port Plug Structure and Diagnostic First Wall for ITER Diagnostic system

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ITER Port Plug will operate at pressures and temperatures which fall under the French Regulation on Pressure Equipment / Nuclear Pressure Equipment. It will be classified as Pressure Equipment. This paper focuses on the application of this regulation to the Equatorial and Upper Port Plug (EPP and UPP) and the constraints induced. These EPP are a size of 3x2x2 m with a maximal weight of 45 T and these UPP are a size of 6x1.5x1.5 m with a maximal weight of 35T. Being close to the plasma, Port Plug structure and Diagnostic First Wall (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. These structures are designated as “Pressure devices” and therefor need to follow French regulation. This paper asses the design of the Port Plug Structure and DFW from a manufacturing, maintenance, safety, planning and design point of view. An analysis of the key load and the relative weighting will be assessed. Optimisation of the design to ensure a fully safe system over the life cycle will be discussed

Id 691

Abstract Final Nr. P3.118

Neutronic analysis of the Diagnostic Equatorial Ports in ITER

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The Diagnostic Port Plugs provide an infrastructure for integrating the diagnostics necessary to operate the machine, while withstanding the nuclear and mechanical loads. They have to provide neutron shielding in order to ensure accessibility to the Port Interspace region for maintenance operations. This a strong design driver for the Equatorial Port Plugs (EPPs) requiring a strong neutron attenuation of 7 orders of magnitude in order to reduce the Shutdown Dose Rates (SDR) in the Port Interspace close to 100 µSv/h after 12 days of shut-down, while maintaining the weight limit. To achieve this challenging task, a series of investigations have been carried out. To mitigate the neutron streaming around the gaps between the Vacuum Vessel Port Extension and the EPP, necessary for its insertion, the design has evolved to an improved double labyrinth configuration with welded shims at the back of the EPP to reduce the gap with the Vacuum Vessel Port Extension. This design proved to be very successful for this task. Neutron stoppers have been places after the gaps necessary for the insertion of the Diagnostic Shielding Modules (DSMs) housed inside the EPP, effectively reducing the neutron streaming inside the EPP. The weight limit reduces the amount of material available for shielding, hence effort was placed in the further development of the DSMs. To accomplish this, the latest design relies on a stainless steel structure and lightweight efficient neutron absorbing materials inside. Studies of cross talk between Ports and the influence of the streaming through the blanket area are presented to give the status of the neutronic environment in the Port Interspace region and their contribution to the SDR. The HELIOS supercomputer and the IO cluster provided the extensive computing power required for the study.

Id 646

Abstract Final Nr. P3.119

Alignment of in-vessel components by metrology defined adaptive machining

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The assembly of ITER will involve the precise and accurate alignment of a large number of components and assemblies in areas where access will often be severely constrained and where process efficiency will be critical. One such area is the inside of the vacuum vessel where several thousand components shall be custom machined to provide the alignment references for in-vessel systems. The abstract gives an overview of the process that will be employed; to survey the intertfaces for approximately 3500 components then define and execute the customisation process.

Id 733

Abstract Final Nr. P3.120

Summary of electromagnetic analyses for the final design of the iter blanket modules

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EM loads are the most challenging for the design of many blanket modules, and, thus, great care must be taken in their assessment. In the last two years, extensive EM analyses of all the most critical ITER blanket modules have been carried out in support of the completion of the Blanket final design. The ITER reference plasma disruptions are numerically simulated using the DINA code on the basis of experimental. The EM transients produced by a plasma disruption can be considered in terms of three separate events: the poloidal field variation (PFV), the toroidal field variation (TFV) and the halo currents (HC) which, while occurring simultaneously, can be analysed separately and superimposed in the analysis post-processing. The DINA outputs provide all the physical information needed to reproduce the equivalent EM transients of these events for the 3D finite-element (FE). Due to the geometric complexity of the blanket modules, a single detailed FE model including all the BMs, together with all electromagnetically relevant structures, would have been by far beyond the limits of the actual numerical analysis capability. The analyses have been thus carried out by preparing different detailed FE models for each blanket module. To reduce the difficult and time-consuming work required to model all the non-BM structures and to check the FE-DINA interfaces for every new analysed BM, an ITER General Model has been developed for EM analyses. Every new FE model has been produced by refining the IGM only in the region of interest while leaving unchanged the plasma region and the main conducting structures surrounding the component under investigation. The following topics will be described: 1) The ITER General Model; 2) The procedures used to interface the IGM with DINA outputs; 3) An overview of the developed meshing techniques; 4) A summary of the main analysis results.

Id 894

Abstract Final Nr. P3.121

Completion of the System Integration of the ITER In Vessel Components

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The design revision phase of the ITER In-Vessel Components, process that started in 2007, shortly after the signing of the ITER treaty, is planned to be finalized during 2014. An integrated design concept has been developed that provides for a coherent design of the in-vessel diagnostics, in-vessel coils, cooling water manifolds, blanket modules and divertor cassettes and their respective support features to the VV. Extensive analysis has been launched to verify that this configuration fulfils the power handling, disruption loads and nuclear shielding requirements Integrated assembly and maintenance schemes have been developed as well. This integrated design has provided the basis to proceed with the detailed design for the different components. Final design reviews are planned or have taken place during 2013 and 2014 for the in-vessel coils, the blanket manifolds, the blankets and the divertor, after which the different manufacturing contracts will be launched by the Domestic Agencies. This paper describes the integrated design that has been chosen, the engineering trade-offs involved, and provides an overview of the detailed designs of the above mentioned components.

Id 759

Abstract Final Nr. P3.122

Status and Challenges in the Design for Remote Handling Compatibility of ITER DNB Components

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For the Neutral Beam systems in ITER, one of the biggest challenges in the design of components and their assembly procedure is that of ensuring Remote Handling (RH) compatibility that conforms to the applicable ITER Remote Handling Code of Practice (IRHCOP). In this paper, the integration of this important RH interface for the Diagnostics Neutral Beam (DNB) system is presented. While, the NB systems in ITER (Heating Neutral Beams (HNB) and DNB) have a similar approach to the RH compatibility, the interface integration needs to follow the mechanical design and its constraints. For DNB, these manifest in optimising the actuator configuration for reduced RH operation and better maintainability of hydraulic interfaces and a customised RH for the integration of the DNB beam Source (DNB-BS) for the horizontal integration with the High Voltage bushing. There is also a specific configuration of a maintainable Caesium oven that is located outside the vacuum enclosure. Further, some of the interfaces in the DNB-BS need to be addressed as Protection Important Class (PIC), implying special incorporation of post maintenance inspection and quality compliance. This paper shall present the following: i) outcome of the RH integration that establishes a feasible routing of the coolant water pipe lines of Beam line components and ES ensuring compatibility for the RH cut/weld activities; ii) design iteration of the interface between the feed lines from HV bushing and the beam source to accommodate various RH tooling; iii) Major design changes in the HV Bushing and Electrostatic Shield for the RH compatibility also discussed; iv) Relocation of the Cs oven from inside the vessel to outside vessel and its RH compatibility , incorporation of demountable joints on the Cs line design has been detailed; v) Design of intermediate adaptors compatible with monorail crane for RH of Beam source and HV Bushing.

Id 65

Abstract Final Nr. P3.125

Software protocol design: communication and control in a multi-task robot machine for ITER vacuum vessel assembly and maintenance

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A specific communication and control protocol for software design of a multi-task robot machine is proposed. For the vessel/in-vessel engineering and remote handling in ITER, the employment of robots has been proposed by many participants and has been proved to be a practical and promising solution for the vessel engineering. However among most of these proposed robotic solutions, the commercial robot is rarely adopted because the commercial robot is designed mostly for a universal usage and is incompetent to satisfy the special requirement in the vessel engineering. To date, the robots proposed in the vessel engineering of ITER are almost new designed, and one common character among those robotics is the multi-functionality. The designers/developers of these robots have to deal with stringent control performance realization which is generally very hardware-specific on one hand, on the other hand have to consider the robotic multi-functionality realization which is more inclined to client/user and task operation. In order to liberate the developers --- especially the software designers, of the robot machine from dealing with complicated bottom hardware-specific design and to permit greater focus on client-oriented function design, a high-level data management protocol is designed, in which all the data passing through the robot is divided into three categories: trajectory-oriented data, task operation-oriented data and status monitoring-oriented data. The protocol consists of three sub-protocols – namely, a trajectory protocol, task control protocol and status protocol, which can be deployed over the Ethernet based on TCP/IP and run as independent processes in both the client and server computer. In the software deployment, the protocol plays the role of middleware, and the software for the robot machine can be separated into three parts: a client-oriented GUI part, protocol part, and control system part. Applying these protocols, software for a multi-task robot machine that is used for ITER vacuum vessel assembly and maintenance has been developed and it is demonstrated that machining tasks of the robot machine, such as milling, drilling, handling and laser scanning, can be realized easily and implemented in both an individual and composite way.

Id 909

Abstract Final Nr. P3.127

preliminary electromagnetic, thermal and mechanical design for fw and vv of fast

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The Fusion Advanced Study Torus (FAST), with its compact Tokamak design (R=1.82m, a=0.64m), high toroidal field and plasma current, faces many of the problems met by ITER and at the same time anticipates much of the DEMO relevant physics and technology. FAST works in an integrated plasma scenario and has as main investigation targets the Plasma Wall Interactions, large ELMs and non-linear dynamic effects of alpha particles. The Vacuum Vessel (VV) is made of Inconel and segmented into 20° modules. The First Wall (FW) is composed of a bundle of poloidal coaxial pipes armoured with 4 mm thick plasma-sprayed W. This configuration is aimed at giving to the structure the required mechanical and thermal capabilities as well as the remote handling features. The conceptual design of the FW and VV has been defined on the basis of FAST operative conditions and of “Snow Flakes” (SF) magnetic topology, which is relevant for DEMO Considerable R&D effort has been spent in the Electromagnetic (EM) design of FW components to make sure that they are capable to withstand the expected electromagnetic loads from the foreseen FAST operative conditions. The EM loads are one of the most critical load components for the FW and the VV during Plasma Disruptions and a first dimensioning of these components for such loads is mandatory. During R&D activities the conceptual design of the FW has been assessed estimating, by means of FE simulations, the EM loads due to a typical Vertical Disruption Event (VDEs) in FAST. EM loads were then transferred on a FE mechanical model of the FAST structures and the mechanical response of the FW design for the analysed VDE event is assessed. Results indicate that design criteria are not fully satisfied by the current design of the VV and FW components; a design up-grade is then needed

Id 830

Abstract Final Nr. P3.128

Pebble bed structures in the vicinity of concave and convex walls

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In present ceramic breeder blankets, both the ceramic breeder material and beryllium are used as pebbles filled in cavities of different geometries. The proper modelling of the thermo-mechanical behaviour of these pebble beds is mandatory for the blanket design. Currently, the pebble beds are considered as homogeneous materials, which is only a first approximation. Previous tomography experiments showed already in detail that structured pebble packing exists close to walls with a different packing factor compared to the bulk, and presumably also different pebble bed features, especially heat transfer characteristics. Such pebble structures have been investigated close to walls of cylindrical containers (concave walls), whereas the objectives of the present work is to include also the case of pebble beds surrounding cylinders (convex walls). This case is relevant to e.g. breeder-out-of-tube blanket concepts, electrical line heaters and thermocouples immersed in the pebble bed. New tomography experiments were performed at the ESRF, Grenoble, France, using aluminium and beryllium spheres of different diameters in cylindrical containers with inner cylinders, again varying the dimensions. Results are presented which show the characteristics of these wall layers as a function of the relevant parameters. One important result is that the increase of the packing factor for the total cavity is related to the increasing regularity and size of the wall regions.

Id 461

Abstract Final Nr. P3.129

On the implementation of new technology models for fusion reactors systems codes

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For the pre-conceptual design of the next generation fusion power plant (DEMO), systems codes are being used from nearly twenty years. In such computational tools, all the reactors components (namely plasma, blanket, magnets, etc.) are integrated in a unique computational system and simulated by means of rather simplified mathematical models. The system code tries to identify the main design parameters (e.g. major radius, net electrical power, toroidal field) and to make the requirements and constraints to be simultaneously accomplished. In fusion applications, requirements and constraints can be either of physics or technology kind. Regarding the latest category, in this study a novel modelling approach is proposed in order to accurately consider the key technology aspects in systems analyses, such as neutronics in the breeding blanket and electromagnetic and mechanical analyses in magnets systems. In particular, Breeding Blanket, Magnets and Fuel Cycle/Vacuum systems are analysed in details, aiming to highlight the connections among these components and the impact on the overall power distribution. As a matter of fact, more accurate technology modules imply better estimations of the key power terms to be considered in systems codes, such as the nuclear power in the breeding blanket and the coolant pumping power. Particular attention is also paid to the geometry models which are necessary for the poloidal representation of the reactor. This study is first characterized by a modelling part, where the proposed new models for Blanket, Magnets and Fuel Cycle/Vacuum systems are briefly described; this part is then followed by a dedicated investigation on methodologies for the integration in systems codes, focusing on the linking parameters between the different technology modules and the plasma physics core. In the final part, the new models are tested on the most up-to-date DEMO version and the preliminary results are presented and discussed.

Id 526

Abstract Final Nr. P3.130

Thermo-mechanical screening tests to qualify beryllium pebble beds with non-spherical pebbles

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In present ceramic breeder blankets, both the ceramic breeder material and beryllium are used in form of pebbles. At present, for beryllium rather spherical pebbles with diameters of about 1mm are considered. Non-spherical particles can be produced much cheaper and are, therefore, of significant economical interest. In contrast to the candidate material (Be-1), adequate pebble bed data basis do not exist for non-spherical beryllium grades, and the blanket relevant potential of these grades cannot be judged. The fundamental classification of granular materials consists of three steps: i) blanket relevant filling experiments for measuring the packing factor , ii) uniaxial compression tests (UCTs) in order to determine the pebble bed stress-strain dependence, iii) the measurement of the pebble bed thermal conductivity k. Experiments were performed with three grades of non-spherical beryllium pebbles produced by Bochvar Institute, Russia, and Materion, USA, accompanied by experiments with Be-1. Because of the limited amounts of the non-spherical materials, pebble bed volumes were relatively small and the experimental set-ups were designed in such a way that general conclusions could be drawn. Previously published filling experiments proved that the packing factor dependence from container dimensions is fairly the same for all grades. Now, UCTs combined with the Hot Wire Technique (HWT) to measure k were performed. A detailed 3d modelling of the experimental set-up was of prime importance in order to prove that the used design was appropriate. In the experiments, first, the uniaxial stress was stepwise increased and at each step strain and k were measured. Then, in a similar way stress was decreased. The Be-1 data agreed well with previously published data. Compared to Be-1, the pebble beds with non-spherical pebbles are characterized by a softer behavior (larger stain value for a given stress value) and smaller increases of k as a function of strain.

Id 508

Abstract Final Nr. P3.131

ITER divertor flow modelling

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The ITER divertor vacuum system has a significant influence on plasma operation and control. The torus vacuum system is mainly used to maintain the required plasma and divertor pressure during the burn phase and to pump out the vessel between the pulses in a limited time. The flow in the divertor region influences the divertor dome pressure and the gas throughput and also the time which is needed to achieve the base pressure between the pulses. The gas passages through the divertor cassettes and torus pumping ducts can be simulated as a 2D network of channels with a known conductance. The asymmetries in the system due to the different shapes, cross sections and lengths of the individual channels as well as the un-symmetric connection of the pumping ducts to the divertor ring contribute to a high complexity. Simulations were performed to study in a parametric way the influential parameters of the gas flow through the divertor and along the pump ducts into the torus cryopumps. For the simulations, the ITERVAC code was used, which covers all flow regimes from laminar flow and transitional flow. Main parameters for the simulations are gas type, temperature, inlet and pump pressures and pumping speeds of the pumps. The results are the difference pressures and related gas throughputs for every channel. This paper presents a complete update of an older, now superseded study [1] of divertor flow modelling and takes into account all divertor system changes that happened since 2007. The results include ITERVAC simulations at different divertor pressures at plasma burn and for the pumping during the dwell phase between pulses. This work has been conducted under the European ITER Physics Support Programme. [1] V. Hauer, Chr. Day, FED 84 (2009) 903-907.

Id 627

Abstract Final Nr. P3.132

Magnetohydrodynamic flow in ducts with discontinuous electrical insulation

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In liquid metal blankets the interaction of the moving breeder with the intense magnetic field that confines the fusion plasma results in significant modifications of the velocity distribution and increased pressure drops compared to corresponding hydrodynamic flows. Those changes are due to the occurrence of electromagnetic forces that slow down the core flow and have to be balanced by large driving pressure heads. The ensuing pressure losses are proportional to the electric current density induced in the fluid, whose magnitude is determined by the resistance of the current path. Therefore additional magnetohydrodynamic (MHD) pressure drops can be minimized by electrically decoupling the wall from the fluid where the large current density is induced. For applications to dual coolant blankets it is foreseen to loosely insert electrically insulating layers into the ducts. In long channels, in order to cover the entire duct wall, the insulation could consist of a number of shorter inserts. This implies a possible local interruption of the insulation. As a result electric currents can additionally close through the conducting duct walls leading to stronger electromagnetic forces that brake the liquid metal flow. Moreover, owing to imperfect insulation provided by the inserts, higher velocities are present in parallel boundary layers which eventually may become unstable. Three dimensional numerical simulations have been performed to investigate MHD flows in electrically well-conducting channels with internal insulating inserts that present a local interruption. A larger pressure drop is observed compared to fully developed conditions and, depending on the flow rate, instabilities may occur in boundary layers along walls parallel to the magnetic field. In the present paper the main flow phenomena are described and additional pressure losses are quantified.

Id 765

Abstract Final Nr. P3.133

Experimental study of instabilities in magnetohydrodynamic boundary layers

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In liquid metal blankets heat and mass transport phenomena are strictly related to velocity profiles near the walls. It is known that the thin boundary layers, which form along walls parallel to an imposed magnetic field, can lose their stability due to the presence of high velocity jets with inflection points. This can lead to the formation of complex vortical structures whose size and features depend on mean flow rate and magnetic field strength. With the aim of investigating experimentally flow instabilities in liquid metal MHD duct flows a loop has been manufactured and fully instrumented, in which Indium Gallium Tin is used as model fluid. The circuit is inserted in a large dipole magnet, whose properties allow reaching flow parameters comparable with those typical for ITER applications. During the experiments electric potential distributions are recorded on the surface of the test section, which consists of a long square duct, and in the liquid metal by means of a movable 4-pole potential probe. Detection of time-dependent signals allows identifying the critical Reynolds number for the onset of instabilities. Time-averaged measured data gives further information about the mean flow distribution and can be directly compared with numerical simulations. The latter ones are used to support the physical interpretation of the observed MHD phenomena.

Id 766

Abstract Final Nr. P3.134

Hot extrusion of Be-Ti powder

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Be-30.8 wt.%Ti powder mixture was extruded in copper and steel containers at 700 and 900 C, respectively. In both cases, achieved extrusion ratio was 7:1. Investigations of microstructure of manufactured Be-Ti rods revealed that processing temperature has a great influence on the metal flow during extrusion and formation of beryllide phases. XRD results proved that brittle intermetallic phases were formed by processing at 900 C, while no evidence of reaction between beryllium and titanium was detected after extrusion at 700 C. Additionally, high-temperature annealing tests of produced Be-Ti samples were performed in order to study the evolution of the phase composition after heat treatment. The effects of different mechanical properties of core materials (beryllium and titanium) and containers on uniform deformation are discussed in this work.

Id 335

Abstract Final Nr. P3.135

Tritium and helium release from highly neutron irradiated titanium beryllide

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Titanium beryllide Be12Ti is considered to be an advanced neutron multiplier material in Helium Cooled Pebble Bed (HCPB) breeding blankets of fusion power plants. Accumulation of helium and tritium in beryllium-containing materials is the result of neutron-induced transmutations. Tritium inventory in Be-Ti materials is very important from the point of view of safety during the operation of the HCPB blankets. To provide a related materials database, a neutron irradiation campaign, HIDOBE-01 experiment, has been performed at HFR in Petten, Netherlands. Up to 3000 and 300 appm of helium and tritium, respectively, were produced in titanium beryllide pellets after irradiation in the temperature range between 600 and 1050 K. Thermal desorption tests using irradiated titanium beryllide specimens were performed with a quadrupole mass-spectrometer (QMS) and an ionization chamber (IC). The gas mixture consisting of high-purity argon with a small additive of hydrogen (Ar+0.1 vol. % H2) was used as a purge gas to transport the released species to the QMS and the IC. Heating rates of 1 and 7 K/min were used by thermal desorption tests up to 1373 K with final exposure for 3 h at the maximum temperature. In this study, titanium beryllide samples having the chemical compositions of Be-5at.%Ti and Be-7at.%Ti were investigated. Whereas only one release peak of tritium was observed during investigations, two peaks corresponding to helium were detected. Compared to pure beryllium, titanium beryllide Be12Ti has an enhanced tritium release what is expressed by lower temperatures relating to release peaks. Investigations of microstructure revealed that irradiated samples consist of at least two phases – pure beryllium and titanium beryllide Be12Ti. Formation of numerous pores was observed, mainly, in beryllium phase during the investigations of Be-Ti specimen irradiated at the highest temperature.

Id 786

Abstract Final Nr. P3.136

Single stage recycling of tritium breeder pebbles by remelting

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Currently, lithium orthosilicate pebbles, with a secondary phase of lithium metatitanate, are being developed as the ceramic tritium breeders for the EU. In order for these pebbles to be proven as a viable material for future fusion reactors, it is necessary to demonstrate the feasibility of both recycling the material as well as enriching them with lithium-6 after burn-up. Due to the nature of the melt based process, it is proposed that the used pebbles can be enriched with lithium by remelting them in the standard process crucible with additions of LiOH. One of the main advantages of the melt-based process route is that no additional wet chemical recycling steps are required. In order to test the ability to reprocess the pebbles using the same method as the initial batch, a large quantity of pebbles were produced and subsequently refilled into the process crucible to produce another batch. This was repeated multiple times, and after each batch, a sample of the produced pebbles was characterised. The main focus was on the accumulation of impurities in the pebbles due to multiple cycles in the platinum alloy crucible. It was seen that the level of impurities initially depends on the impurities in the raw materials and thereafter a moderate increase in Pt and Rh levels in the remelted material. After validating the ability to simply remelt the pebbles in the standard crucible, pebbles with a simulated end-of-life burn-up of Lithium-6, which is expected for DEMO, were produced. It was assumed that a 50% 6Li-enriched material will be used and that a burn-up of 15% will occur. Finally it was shown that the pebbles produced with depleted lithium levels could be enriched by remelting them with corresponding quantities of LiOH to gain the composition of the intended starting material.

Id 787

Abstract Final Nr. P3.137

Activities for the calibration of the TRIMO++ software in view of simulation of the ITER WDS-ISS configuration

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The development of the configurations of the ITER Water Detritiation System (WDS) and the Isotope Separation System (ISS) and the detailed design of both systems shall use specific software that complies with ITER requirements for the software related to safety issues. Since the Liquid Phase Catalytic Exchange (LPCE) process of the WDS system is the final barrier against tritium release into the environment, the design of this process shall be based on proven software for the ITER relevant operation conditions. The paper aims to provide detailed comparison between the calculated performances of the LPCE process and the measurements collected during many years of the operation of the LPCE process of TRENTA facility from the Tritium Laboratory. The operation conditions of TRENTA facility are relevant for ITER WDS such as deuterium, tritium concentrations and superficial velocity of the fluids phases involved in the process. Evaluation of the requirements of software that shall simulate the cryogenic distillation process and the strategy for calibration against experimental measurements is provided as well. Keywords: water detritiation, liquid phase catalytic exchange, cryogenic distillation

Id 794

Abstract Final Nr. P3.138

Experimental study of permeation and selectivity of different inorganic membranes for tritium processes

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Zeolites are known as tritium compatible inorganic materials widely used in packed beds as driers in detritiation systems and also suggested for tritium removal from helium at cryogenic temperature. Tritium Laboratory Karlsruhe (TLK) has proposed fully continuous approach for tritium extraction from the breeding blanket of fusion machines, based on membrane permeation as a pre-concentration stage upstream of a final tritium recovery by a catalytic Pd-based membrane reactor. Zeolite membranes have been identified as the most promising candidates to improve the overall tritium management minimizing both the tritium inventory and processing time. Experimental studies exploring the permeation properties of zeolite membranes, necessary for future engineering and design work, have been started using the dedicated ZIMT (Zeolite Inorganic Membrane for Tritium) facility. Firstly, a prototype composite MFI-alumina hollow fibre (produced by IRCELYON France) was characterised for single gases (H2, He), binary (H2 + He) and ternary (H2 + He + H2O) mixtures. This membrane has exhibited high permeances 10 m3/m2 h bar for H2 at room temperature. Limited H2/He selectivity not exceeding 2.2 and nearly independent from binary mixture composition was measured. For ternary mixtures, water was almost the exclusive species permeating through the membrane at room temperature. Presently, new commercially available zeolite membranes of different types (ZSM-5, S-SOD, NaA) and one carbon membrane (produced by Fraunhofer Institute of Ceramic Technologies and Systems, Hermsdorf, Germany) are being tested. Parametric study covers the operating temperature and the composition of the inlet mixtures. The first tests with single gases in ZSM-5 membrane showed comparable performances to MFI hollow fibre ones 6 m3/m2 h bar for H2 at room temperature) at the same selelctivity level. This paper will detail the experimental results obtained on the different membranes revealing the most promising candidates for future tests with tritium.

Id 517

Abstract Final Nr. P3.139

Experimental characterisation of random packing materials for cryogenic hydrogen isotope separation

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The chosen method of hydrogen isotope separation in the ITER isotope separation system (ISS) is cryogenic distillation (CD). For a CD separation column, the selection of an appropriate packing material has a major impact on ISS performance, especially separation performance and tritium inventory. While the height equivalent of a theoretical plate (HETP) is the key figure for the separation performance, the liquid holdup will dictate the tritium inventory for a separation column of a given length and diameter. The TRENTA facility at the Tritium Laboratory Karlsruhe (TLK) was modified accordingly, to allow measurement of HETP and liquid holdup using different packing materials and different column dimensions. The paper will give an overview of the modifications to the facility and investigations related to key performance parameter of different random packing material in correlation with column diameter, column length and packing surface treatment. Furthermore, these results will be compared with existing literature.

Id 897

Abstract Final Nr. P3.140

Experimental study at large flow rate of a technical scale catalytic membrane reactor for its potential use for tritium processing in the breeding blanket

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The PERMCAT process relying on counter current isotopic swamping in a catalytic Pd-membrane reactor proved to be efficient and versatile. PERMCAT is particularly attractive since it recovers tritium from different species in a single stage, and produces a pure molecular tritium stream easy to reuse in the fuel cycle. Originally developed for the tokamak exhaust processing, PERMCAT has been recently proposed for recovering the tritium from the purge gas of solid breeding blanket. However, the flow rates to be processed are much larger making the size of the reactor very challenging. A deeper knowledge and better understanding of the limiting factors at high gas flow velocity is required to optimize the process and reactor for this particular application. A dedicated experimental campaign is performed using the latest and biggest PERMCAT reactor produced at the Tritium Laboratory Karlsruhe. It comprises two identical units placed in series, each consisting in 13 Pd-Ag membranes inserted in a single Ni-based catalyst bed. The feed stream contains D2O vapour in He carrier gas, and H2 is used as purge gas to deplete the deuterium in the feed water. Large ranges of operating conditions are being studied varying all the key parameters such as the three flow rates, the pressures at both sides of the membranes, and the temperature of the reactor. A significant and fast decrease of the decontamination factors from 800 down to 6 was measured when increasing the feed flow rate from 0.5 up to 5 L/h. Increasing favourable parameters like feed pressure, hydrogen swamping ratio and temperature can compensate to some extent. Cleary elucidating the limiting mechanisms suspected to be the isotopic exchange kinetics will be possible having a meticulous comparison of these experimental results with numerical predictions for which the simulation tool still under development is being improved in parallel.

Id 951

Abstract Final Nr. P3.142

Manufacturing pre-qualification of a Short Breeder Unit mockup (SHOBU) as part of the roadmap towards the out-of-pile validation of a full scale Helium Cooled Pebble Bed Breeder Unit

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The key components of the Helium Cooled Pebble Bed Test Blanket Module (HCPB TBM) in ITER are the Breeder Units (BU). These are the responsible for the tritium breeding and part of the heat extraction in the HCPB TBM. After a detailed design and engineering phase performed during the last years in the Karlsruhe Institute of Technology (KIT), a reference model for the manufacturing of a HCPB BU mock-up has been obtained. The mid-term goal is the out-of-pile qualification of the thermal and thermo-mechanical performance of a full-scale HCPB BU mock-up in a dedicated helium loop. Several key manufacturing technologies have been developed for the fabrication of some parts of the BU. In order to pre-qualify these techniques, a SHOrt Breeder Unit mock-up (SHOBU) has been constructed and tested. This paper aims at describing the constitutive parts of SHOBU, the manufacturing technologies involved, the assembly sequence (taking into consideration functional required steps like its filling with Li4SiO4 pebbles or its assembly in the HCPB TBM) and the welding procedures executed. The paper concludes with a description of the required pre-qualification tests performed to SHOBU, i.e. pressure and leak tightness tests, according to the standards and with a final exposition of the relevance of SHOBU with a full-scale HCPB BU.

Id 983

Abstract Final Nr. P3.143

Spreading behavior of distributions of hydrogen isotopes adsorbed in zeolite packed bed at 77.4 K

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We have been developing a hydrogen isotope separation system using pressure swing adsorption (PSA) process with synthetic zeolite packed columns at 77.4 K, of which a candidate design schemes to make its work consecutively by operating twin or triplet columns in the manner of a merry-go-round. One cycle PSA process operation consists of (i) isotope separation by displacement adsorption of isotopic mixture for pure H2 as carrier pre-adsorbed in a packed column, (ii) recovery of heavier isotope enriched hydrogen by evacuating the column and then (iii) replenishing the column with the carrier for flushing back the residual adsorbed mixture and pushing back its distribution toward the column's inllet. These steps are alternatively synchronized but the time required for each process is not necessarily same. So, a waiting period for the next step is produced in a step other than that of the longest process (corresponding to Step (ii)). While in the waiting mode the columns are closed and resting, adsorbate molecules are moving inside these columns though the operational flow is stopped. This molecular behavior makes a decline in efficiency of separating the heavier isotopes from the carrier H2 flowing out from a column in the adsorption mode (i), because isotope concentration distributions formed in the packed bed are spreading also during the period of operation in repose. Thus, in this work, the effect of opreational rest time on the concentration distribution spreading was investigated experimentally by observing the breakthrough curves of tracer D2 in a H2-D2 mixture with synthetic zeolite packed columns at 77.4 K, which were obtained by go-stop-go mode operations, and were interprited by simulating the experimental curves with the numerical curves from an analytical mass-transfer model of displacement adsorption of H2 and D2 in adsorbent packed bed.

Id 940

Abstract Final Nr. P3.144

Dual temperature dual pressure water-hydrogen chemical exchange for water detritiation

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Detritiation of liquid effluents from a fusion reactor system is necessary to prevent releases of tritium into the environment. The amount of tritium handled in the system increases with the size of the reactor and detritiation systems also have possibility to enlarge impractically. In the case of a water detritiation system one of the most indispensable processes is a Combined Electrolysis and Catalytic Exchange (CECE). In the process a very long reactor column having huge diameter will be used and especially an electrolyzer is destined to have a heavy burden. As a consequent it is necessary to develop a more effective column and/or a process reducing the burden on the electrolyzer in order to realize an economic and feasible water detritiation system for future fusion reactors. In the present paper we propose a dual temperature dual pressure water-hydrogen chemical exchange (DTDPCE) process as a pretreatment system of the CECE. Experimental and analytical studies on hydrogen-tritium isotope separation by a DTDPCE with liquid phase chemical exchange columns have been carried out in order to apply it to the water detritiation system for fusion reactors. Kogel catalysts and Dixon gauze rings were mixed at a certain ratio and packed in the column in a random manner. Performance tests of tritium separation by the system were performed at various combinations of pressures and temperature. Total separation factors of the apparatus were also predicted by the channeling satage model and compared with the experimental value.

Id 197

Abstract Final Nr. P3.145

Pre-conceptual Design Study on K-DEMO Ceramic Breeder Blanket

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As a following step of the Fusion Energy Development Promotion Law (FEDPL) enacted in 2007 in Korea, a pre-conceptual design study for the Korean fusion demonstration tokamak reactor (K-DEMO) has been initiated with the uniqueness of high magnetic field (BT0 = 7.4 T), major and minor radii of 6.8 m and 2.1 m, and steady-state operation. The main functions of K-DEMO blanket are to absorb radiation and heat loads from plasma, to provide neutron shielding to the components, and produce the self-sustainable tritium by achieving a global TBR > 1.05. A water-cooled ceramic breeder blanket is assumed as the primary candidate of K-DEMO blanket with the following materials: Tungsten as the first wall, Li4SiO4 as breeder, beryllide as multiplier, B4C as shield, ferritic-martensitic steel (RAFM) as structure, and water as coolant. The blanket modules are toroidally divided into 16 inboard and 32 outboard sectors, respectively. Inboard and outboard sectors are further divided into 8 and 10 unit modules, respectively. Each unit module consists of breeding and shielding zones in radial direction. An equatorial port opening, occupying approximately 1.5 m2 and surrounded laterally by neighboring blanket modules, is located at around middle of each 22.5 degree of toroidal region. The upper and lower divertors are located at both vertical ends of the blanket modules. The blanket modules were designed to accommodate the peak thermal and neutron loads on each unit module are ~0.5 MW/m2 and ~3.0 MW/m2, respectively. The mechanical, thermo-hydraulic analyses by using ANSYS have been carried out to support the developed blanket concept. The detailed layer structure of blanket unit module was optimized by MCNP calculations.

Id 189

Abstract Final Nr. P3.146

Electromagnetic analysis on Korean Helium Cooled Ceramic Reflector(HCCR) TBM during plasma major disruption

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Korean Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM) will be installed at the #18 equatorial port of the Vaccum Vessel in order to test the feasibility of the breeding blanket performance for forthcoming fusion power plant in the ITER TBM Program. Since ITER tokamak contains Vaccum Vessel and set of electromagnetic coils, the TBM as well as other components is greatly influenced by magnetic field generated by these coils. By the electromagnetic(EM) fast transient events such as major disruption(MD), vertical displacement event(VDE) or magnet fast discharge(MFD) occured in tokamak system, the eddy current can be induced eventually in the conducting components. As a result, the magnetic field and induced eddy current produce extremely huge EM load(force and moment) on the TBM. Therefore, EM load calculation is one of the most important analysis for optimized design of TBM. In this study, a 20-degree sector model for tokamak system including central solenoid(CS) coil, poloidal field(PF) coil, toroidal field(TF) coil, vaccum vessel, shield blankets and TBM set(TBM, TBM key, TBM shield, TBM frame) is prepared for analysis by ANSYS-EMAG tool. Concerning the installation location of the TBM, a major disruption scenario is particularly applied for fast transient analysis. The final goal of this study is to evaluate the EM load on HCCR TBM during plasma major disruption.

Id 77

Abstract Final Nr. P3.147

Preliminary failure modes and effects analysis on Korean Helium Cooled Ceramic Reflector TBS to be tested in ITER

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Korean Helium Cooled Ceramic Reflector (HCCR) Test Blanket System (TBS), which comprises Test Blanket Module (TBM) and ancillary systems in various locations of ITER building, is operated at high temperature and pressure with decay heat. Therefore, safety is utmost concern in design process and it is required to demonstrate that the HCCR TBS is designed to comply with the safety requirements and guidelines of ITER. Due to complexity of the system with many interfaces with ITER, a systematic approach is necessary for safety analysis. This paper presents preliminary Failure Modes and Effects Analysis (FMEA) study performed for the HCCR TBS. FMEA is a systematic methodology in which failure modes for components in the system and their consequences are studied from the bottom-up. Over eighty failure modes have been investigated on the HCCR TBS. The failure modes that have similar consequences are grouped as Postulated Initiating Events (PIEs) and total nine reference accident scenarios are derived from FMEA study for deterministic accident analysis. Failure modes not covered here due to evolving design of the HCCR TBS and uncertainty in maintenance procedures will be studied further in near future.

Id 854

Abstract Final Nr. P3.148

Experimental evaluation for fuel delivery in the ITER Tritium Plant

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The Storage and Delivery System (SDS) of the ITER Tritium Plant has to safely handle fuel gases including large quantities of tritium-contaminated gases. The SDS shall have sufficient capacity in the inventory amount and flowrate to deliver fuel gases to the Fuelling System (FS): high concentrated tritium and tritium-contaminated deuterium gases. Based on limited number of tritium compatible equipment, however, wide ranges of throughput with highly leak tightness are required to handle fuel gases in the SDS. The optimal process design of the SDS shall be developed under the evaluation of some configurations of equipment: vessels, tritium compatible pumps, tritium storage beds, etc. In this study, experimental evaluation of the fuel delivery will be performed to satisfy the required fuelling rates under the inventory limitation of the tritium-contaminated gases in a safety-manner. The supply of fuel gases to the FS and the recovery of those from Isotope Separation System (ISS) will be experimentally simulated in detail from an interface point of view between the SDS and those systems. Various proposed process concepts to deliver the fuel gases will be evaluated with respect to limited operating conditions: the number of pumps, the number of vessels, and upper/lower pressure operating limits. Control logic will be developed to satisfy the operating requirements of the pressure condition in the fuelling lines between the SDS and the FS. The experimental results will support the decision making to determine the process design of the SDS.

Id 328

Abstract Final Nr. P3.149

Feasibility study of fusion breeding blanket concept employing graphite reflector

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To obtain high tritium breeding performance with limited blanket thickness, most of solid breeder blanket concepts employ a combination of lithium ceramic as a breeder and beryllium as a multiplier. In this case, considering that huge amount of beryllium are needed in fusion power plants, its handling difficultiy and price can be a major factor to be considered for commercial use. Korea has proposed a Helium-Cooled Ceramic Reflector (HCCR) breeding blanket concept for Test Blanket Module (TBM) program which aims to test and validate the concept relevant to DEMO and/or fusion power plants. Here, graphite is considered as a reflector material for neutron irradiation. It is unique feature among fusion breeding blankets that the amount of beryllium is reduced up to 50% by replacing with graphite satisfying the tritium self-sufficiency condition. However, there are still concerns on the graphite reflector related to the nuclear performance as well as material-related issues. In this paper, a nuclear analysis is performed under the fusion reactor condition to address the feasibility of graphite reflector in breeding blanket, considering tritium breeding capability and neutron shielding and activation aspects. Also, the chemical stability of the graphite is investigated considering the possibility of chemical excursion under explosion condition, resulting in that the adaptation of graphite reflector in breeding blanket is intrinsically safe under fusion reactor condition.

Id 971

Abstract Final Nr. P3.150

Hydrogen solublity in FLiNaK mixed with metal powder

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Fluoride molten salt (FLiNaK, FliBe etc.) is attractive tritium breeding material with low electric conductivity, low reactivity with air or water and good compatibility with structure material. However, one great concern on using molten salt as a liquid breeder is its very low hydrogen solubility which increases the risk of tritium leakage. A. Sagara has proposed that mixing molten salt with metal powder with high hydrogen solubility can increase the effective hydrogen solubility keeping the benefits of molten salt intact. In this work, FLiNaK (LiF-NaF-KF (46.5-11.5-42 mol %)mixture) was mixed with powder of Ti or Zr. The mixture was contained in Ni crucible and melted, followed by hydrogen gas feeding in a tube furnace to quantify the effective solubility. In case of Zr powder mixture (Zr powder(45µm grain): 160mg, FLiNaK 15.0g, Ni crucible: 32.0g)at 823 K, Zr powder was estimated to absorb hydrogen up to 0.05 in H/Zr molar ratio within 10 hours when the ambient hydrogen gas pressure was around 1.2 kPa. The value is not so high as the one calculated from the ideal absorbing capacity (H/Zr ~1.0) of the Zr powder, however the effective solubility of the mixture is larger than that of pure FLiNaK for almost 3 orders of magnitude. According to these results, mixing of molten salt with metal powder is shown to be hopeful to enhance the effective hydrogen solubility and to reduce the risk of tritium leakage.

Id 982

Abstract Final Nr. P3.151

Calculation code 'TC-FNS' for a DT fuel cycle of a steady-state fusion reactor

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Technology of fusion DT fuel cycle is one of key elements for a next-step steady-state tokamak like DEMO or fusion neutron source (FNS). This technology should be significantly developed because technical solutions selected in the ITER project may be used in such devices only partially due to higher capacity factor for a steady-state facilities, high neutron fluxes, temperature and tritium flows in the fuel cycle elements. To estimate the distribution of tritium in the systems of a fusion facility and elements of 'tritium plant' a code “TC-FNS” was developed. The code allows to carry out calculation of the fuel flows and inventory in the systems of tokamak like vacuum vessel (VV), cryopumps, NBI system, purification system, isotope separation system and tritium storage system. The input parameters for the code are the parameters of tokamak and sub-systems (geometric dimensions of the VV, the plasma parameters (particle confinement time, plasma density, etc.), retention by materials of VV, energy and current of NBI, operating regime of cryopumps , etc.). The code take into account the mechanisms of reduction of tritium in the fuel cycle due to thermonuclear burn and β-decay in all systems. In the paper with using “TC-FNS” code describes a concept of DT-fuel cycle for a steady-state FNS based on tokamak. The calculation results for the fuel cycle DEMO-FNS (R= 2.5 m, aspect ratio A = 2.5, 500 keV NBI (30 MW), 30-40 MW of fusion power) compared with calculations with fuel cycle for FNS-ST (R= 0.5 m, aspect ratio A = 1.66, 130 keV NBI (10 MW), 2-3 MW of fusion power) and for the ITER scale facilities will be presented.

Id 586

Abstract Final Nr. P3.152

Rapid material development and processing of complex shaped parts via tungsten powder injection molding

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The advantages of tungsten, for example the high strength and thermal conductivity, the low thermal expansion, low tritium inventory and low erosion rate make its attractive to be used in a wide range of applications for fusion power plants. Unfortunately, the commercial fabrication route is suitable for a narrow range of alloys only. In addition, the semi-finished products are limited to plates and rods. Therefore, at Karlsruhe Institute of Technology the mass production of near-net shaped tungsten parts are intensively investigated. Here, Powder Injection Molding (PIM) is the ideal tool for an easy fabrication of complex shaped components, the joining of different materials without brazing, and a rapid development of new tungsten materials. The focus of this contribution is laid on the development, fabrication and study of various new tungsten PIM materials. The process chain is briefly discussed and includes the design and engineering of PIM tools as well as the required filling simulations. This enables the replication of complex parts in FUSION relevant dimensions. Moreover, it is shown that the material properties can be varied in a wide range by adopting the process parameters. In this context, microstructural and mechanical characterization results demonstrate the successful approach.

Id 51

Abstract Final Nr. P3.153

Consolidation process studies for Ferritic ODS steels

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Oxide Dispersion Strenghtening (ODS) is used to produce advanced steel grades which are potential structural materials for fusion application. In this study a ferritic premix powder (Fe-13Cr-1W-0.3Ti) was mechanically alloyed in an attritor ball mill with some Fe2Y intermetallic powder to produce ODS steel. Several processes can be used to consolidate the resulting ODS powder. In this study we investigated Spark Plasma Sintering (SPS) and Hot Isostatic Pressing (HIP) followed by hot rolling. The processing parameters were varied in order to determine their influence on the resuting densities, microstructure and mechanical properties. Then a comparative assessment of the different consolidation processes was performed.

Id 304

Abstract Final Nr. P3.154

Needs and gaps in the development of aluminum-based corrosion and T-permeation barriers for DEMO blankets

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Low-activation-ferritic-martensitic (RAFM) steels are candidates as structural materials in HCLL, WCLL and DCLL blanket development for DEMO and partly for TBM’s tested in ITER. All these designs have in common that the liquid breeder Pb-15.7Li is in direct contact with the structural material and thus two mayor topics, corrosion and T-permeation, will gather the reliable, safe and economical application of such a combination of breeder and structural material. Research activities on corrosion properties in flowing Pb-15.7Li of such RAFM steels, e.g. Eurofer, revealed strong dissolution attack and partly dramatic corrosion rates of 400 µm/a at 550°C and flow velocities of 0.22 m/s. These findings necessitated the development of corrosion barriers to protect these steels against corrosion and therefore different coating processes where under development to produce aluminum-based barriers in the last years. Up to date, coatings made by hot-dip aluminizing (HDA), and by electrochemical deposition (ECA, ECX process), proved their ability to protect Eurofer against corrosion in flowing Pb-15.7Li. Besides these corrosion related issues, these coatings are envisaged also to act as T-permeation barriers to reduce/avoid T-permeation from the liquid breeder into the coolant, but available T-permeation data for coated RAFM steels, in general, are rare and such qualification is outstanding especially for barriers made by ECX process. This paper summarizes the state-of-the-art of aluminum-based barrier development and clearly points out the gaps and needs in future scale development / characterization and T-permeation barrier development for application as DEMO blankets. Additionally, necessary qualification steps are outlined on the path towards a reliable fabrication route to produce aluminum-based corrosion and T-permeation barriers on RAFM steels for blanket applications in future fusion reactors.

Id 358

Abstract Final Nr. P3.155

Shieling design optimization for the IFMIF test facility based on high-fidelity Monte Carlo neutronic analyses

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The IFMIF (International Fusion Material Irradiation Facility) project is in the so-called EVEDA phase and the Intermediate IFMIF Engineering Design Report (IIEDR) was delivered in June 2013. The test cell (TC) is the central part of IFMIF where an intensive neutron field is created by d-Li nuclear reactions and the test modules (TMs) are placed to irradiate candidate fusion reactor materials. The present study is devoted to further investigations of two open issues on the reference TC design addressed in the IIEDR by high-fidelity neutronic analyses with state-of-the-art tools. An important feature of the reference TC design is that the all pipe and cable penetrations for the TMs are separated from shielding structures and accommodated by removable piping and cabling plugs (PCPs). The neutron streaming effect along the cable/pipe penetrations and existing gaps around the removable shielding plugs is crucial to justifying the present TC design. Another issue is the optimization of the cooling pipe arrangement in the shielding structure. To this end, a very detailed geometrical model for neutronic analyses has been prepared directly from engineering CAD data by utilizing the McCad conversion software developed at KIT. All the removable plugs are separately described in the geometrical model and a PCP model including pipes was incorporated. The calculation result suggests that the streaming contribution is restrained if the penetrations and gaps are designed appropriately, and the reference TC design has been successfully validated. The same geometrical model was utilized to calculate a detailed 3-dimensional nuclear heating distribution. The complicated mesh-based nuclear heating data have been transferred to an unstructured mesh for a fluid-dynamics simulation by using the McMeshTran tool developed at KIT without deteriorating its accuracy. The present study shows that these tools are very effective measures for such a complex fusion facility design.

Id 393

Abstract Final Nr. P3.156

Determination of RAFM steel properties at high temperatures by instrumented indentation

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For the design and development of structural materials assigned for the use in future fusion reactors, the prediction of their mechanical behavior after irradiation is indispensable. For this purpose, instrumented indentation is an attractive method for testing of even small neutron-irradiated specimens. From the continuously recorded indentation depth and the indentation force, it is possible to deduce mechanical parameters of the tested material i.e. material´s hardness, Young´s modulus and yield point. With a high-temperature indentation device, developed at KIT, it is possible to test materials at temperatures up to 650 °C under remote-handling conditions. The use of an appropriate material for the indentation tip with a stable behavior at high temperatures is an additional challenge. Besides the investigation of the material behavior, the influences of testing-temperatures and forces on the indentation tips are analyzed, regarding the mechanical and chemical resistance. For the present investigations a diamond tip and a sapphire tip were used. For the first time the material EUROFER, a reduced-activation ferritic-martensitic steel for fusion application, is characterized by indentation in his full range of application up to 500°C after different heat treatments. A comparison with the data of annealed tensile tests of the same materials shows that the hardness is highly depending on the testing-temperature and can be correlated to the tensile results. Besides EUROFER, the material MANETII is also investigated in the present study to further analyze to what extent the heating of the samples during the indentation and the corresponding holding times have an influence on the mechanical behavior of the material. This is tested with indentations at room temperature before and after the high-temperature tests. The results of the present study show the functionality of the high-temperature indentation device. The study builds the foundation for future investigations of highly activated irradiated EUROFER-samples.

Id 731

Abstract Final Nr. P3.157

Comparative assessment of different approaches for the use of CAD geometry in Monte Carlo transport calculations

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Computer aided design (CAD) is an important industrial way to produce high quality designs. Therefore, CAD geometries are in general used for engineering and the design of complex facilities like the ITER tokamak. Although Monte Carlo codes like MCNP are well suited to handle the complex 3D geometry of ITER for transport calculations, they rely on their own geometry description and are in general not able to directly use the CAD geometry. In this paper, three different approaches for the use of CAD geometries with MCNP calculations are investigated and assessed with regard to calculation performance and user-friendliness. The first method is the conversion of the CAD geometry into MCNP geometry employing the conversion software McCad developed by KIT. The second approach utilizes the MCNP6 mesh geometry feature for the particle tracking and relies on the conversion of the CAD geometry into a mesh model. The third method employs DAGMC, developed by the University of Wisonsin-Madison, for the direct particle tracking on the CAD geometry using a patched version of MCNP. These three very different approaches are analysed to identify advantages and disadvantages of each method in generating and using a complex geometry model in Monte Carlo neutron calculations. The assessment is performed on the basis of an ITER benchmark model following each approach through the entire simulation process; starting from the same CAD geometry and calculating the same nuclear responses. The usability of each method is assessed by comparing the model preparation time, the computation time for the same number of histories and the accuracy of the obtained results. For each approach the findings are discussed and, as far as possible, quantified with respect to user-friendliness and calculation performance. The obtained results show that each method has its advantages depending on the complexity and size of the model, the calculation problem considered, and the expertise of the user.

Id 820

Abstract Final Nr. P3.158

Microstructural anisotropy of ferritic ODS alloys after different production routes

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Concepts for future generation fusion reactors have demanding requirements for the structural materials. High neutron doses and elevated temperatures form a harsh environment in which present commercially available materials cannot be used. A new class of oxide dispersion strengthened (ODS-) materials which are currently being developed have a high chance of meeting these requirements. The application of ODS steels as functional or structural application strongly depends on the availability of large batches of materials. Since no commercial ODS-alloys are available at the moment, investigations on large scale batches are crucial for future applications. In this study, a batch of 10 kg ferritic steel powder (Fe-13Cr-1W-0.3Ti) was mechanically alloyed in a semi-industrial attritor ball mill with Fe2Y intermetallic powder. Batches of 3-4 kg with different powder particle size distributions were canned in stainless steel containers and compacted by hot-isostatic-pressing (HIP) and hot-extrusion. A thermo-mechanical treatment including hot rolling and annealing was performed for the as-HIPed alloys afterwards. The effects of powder particle size distribution on the microstructural properties were studied by scanning electron microscopy with electron backscatter diffraction. Although a homogeneous and fine grain size distribution was achieved after rolling, areas with different amounts of deformation were found. Microstructural properties and anisotropy effects are correlated to Charpy-impact tests. The materials after rolling shows a DBBT well below -50°C with excellent impact energies. Detailed investigations on the crystallographic textures concluded the work.

Id 831

Abstract Final Nr. P3.159

Tensile Behavior of EUROFER ODS Steel after Neutron Irradiation up to 16.3 dpa between 250 and 450 °C

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For nuclear use, it is always the aim to develop an irradiation tolerant steel with a high strength and a high ductility. Two 50 kg heats of a reduced activation ferritic-martensitic 9%CrWVTa steel with nanoscaled Y2O3-particles, EUROFER-ODS, have been produced using power metallugy fabrication technology. One important step during this optimizing work was an irradiation program up to 16.3 dpa, 771 FPD, between 250 and 450 °C, in the High Flux Reactor (HFR), The Netherlands. The first material of EUROFER97 ODS HIP (0.5% Y2O3) was included and prepared for a post-irradiation tensile test program. The tensile behavior was compared with the tensile properties of steel EUROFER97. All these specimens showed the highest tensile strength at lower irradiation temperatures 250 – 350 °C. But, there was a clear difference in the mechanical behavior between steel and ODS-alloy, which could be documented by fully instrumented tensile tests. First, in the unirradiated state, there was already a considerable increase of about 60% in tensile strength of the ODS-alloy in contrast to the steel. This strengthening was 20% further increasing during the neutron irradiation, - but with a much better ductility than observed in the steel. The typical irradiation induced strain localization of EUROFER97 or RAFM steels could not be observed in the EUROFER97 ODS HIP alloy. All results of post-irradiation tensile tests of EUROFER97 and EUROFER97 ODS HIP will be presented and the opposed material behavior discussed together with fractographic observations.

Id 747

Abstract Final Nr. P3.160

Low temperature mechanical properties of soft solders for superconducting applications

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Electrical connection of the superconductors with each other or with a parallel shunt metal are important part of every superconducting system. Soft soldering is commonly used for the electrical interconnection of superconducting strands. However, soldering of these electrical interconnections should be carried out at temperatures lower than 200° C, in order to prevent a degradation of the superconducting material and avoid oxygen depletion of the superconductor. Therefore, solders with low melting temperatures are of special interest for superconducting applications. For the design of these electrical joints, along with electrical properties the mechanical properties of solders at low temperatures are important to know. Usually the materials change their mechanical properties significantly when going to low temperatures, which can decrease the quality of the joint or even completely destroy it. In specific cases joints have to be able to withstand mechanical loads during operation of a superconducting device. Therefore, it is challenging task to identify a soft solder meeting all specific requirements. In this work low temperature mechanical properties of several soft solders, which are commonly used for cryogenic applications, are presented.

Id 936

Abstract Final Nr. P3.161

Overview of IFMIF EVEDA Test Facility Design

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IFMIF (International Fusion Material Irradiation Facility) is an accelerator based intense neutron source for studying and qualifying structural and functional materials for DEMO and future fusion nuclear power plants. In the current EVEDA (Engineering Validation and Engineering Design Activities) phase, engineering design of the test facility (TF), one of the three key facilities of IFMIF, has been conducted under the European-Japanese “Broader Approach” framework. The TF includes the systems required to accommodate the Test Modules (TMs) at a controlled environment and conditions for irradiation as well as all the systems required for assembling and disassembling the TMs and the equipments for sending the irradiated specimens to post irradiation facility. In the current IFMIF EVEDA TF design, the TF comprises a series of TMs, in which material specimens are installed, one Test Cell (TC), which accommodates all TMs and part of lithium target system for irradiation experiments, one access cell, where remote handling systems are installed to transport TMs and to perform maintenances in the TC, four test module handling cells for processing irradiated TMs and assembling TMs, and test facility ancillary systems, which provide media and power to other TF systems and receive/process signals. In this paper, major functions and specifications of the TF and the above mentioned systems are outlined, the basic configuration of the TF is described, and the current status of the engineering design of key components is overviewed.

Id 926

Abstract Final Nr. P3.162

Friction stir welding, a possible method to join reduced activation structural materials like eurofer and eurofer-ods steels?

Rainer Lindau (1), Michael Klimenkov (1), Ute Jäntsch (1), Anton Möslang (1), Luciano Bergmann (2), Jorge dos Santos (2),

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Friction Stir Welding, a possible method to join reduced activation structural materials like Eurofer and Eurofer-ODS steels? R.Lindau 1, M. Klimenkov1, U. Jäntsch1, A. Möslang1, L. Bergmann2 , J.F. dos Santos2 1Karlsruhe Institute of Technology, Institute for Applied Materials, Karlsruhe, Germany 2Helmholtz-Zentrum Geesthacht, Institute of Materials Research, Solid State Joining Processes, Geesthacht, Germany The use of reduced activation ferritic martensitic (RAFM) oxide dispersion strengthened (ODS) steels like the 9%Cr Eurofer-ODS and higher chromium containing ferritic ODS steels as structural materials in blanket as well as divertor applications of advanced future power nuclear fusion reactors would allow to increase the operational temperature to 650-750°C compared to standard Eurofer, the European reference steel for DEMO structures. One drawback of ODS steels is difficult joining. Standard fusion welding techniques can only be applied in regions of lower demands since the melting process leads to a loss of strength due to coarsening and agglomeration of the strengthening nanometric ODS particles. Friction stir welding (FSW), a solid-state joining process, is considered as alternative method to join ODS alloys. Within this work similar and dissimilar joints of Eurofer-ODS and Eurofer have been fabricated by FSW. The microstructure of different areas of the welds (un-affected parent material, weld nugget and heat-affected zone) were characterized by metallography, scanning electron microscopy (SEM), electron back scattered diffraction (EBSD), and analytical transmission electron microscopy using a 200 kV Tecnai 20 FEG TEM. Vickers micro-hardness measurements were used to determine the hardness changes from the un-affected base material over the heat-affected and stir zone. Eurofer and Eurofer-ODS show a totally different behaviour compared to the higher chromium containing ODS steel PM2000 which was investigated for comparison. This can be attributed to the high frictional heat input during the FSW process and the different response of the ferritic martensitic and purely ferritic alloys. Tensile tests on micro-tensile specimens at room temperature and elevated temperatures (500, 700°C) were performed to assess the tensile properties of the welding zone and base material with and without post-weld heat treatment (PWHT). The tensile behaviour can be well correlated with the micro- and nanostructural changes introduced during the FSW process.

Id 996

Abstract Final Nr. P3.164

Microstructure evolution and impact toughness in the weld heat-affected zone of a reduced activation 9Cr-2W-VTa ferritic/martensitic steel

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Microstructure and impact properties in the weld heat-affected zone (HAZ) of a reduced activation 9Cr-2W-VTa ferritic/martensitic steel were explored through transmission electron microscopy (TEM) analysis and Charpy V-notch impact tests. The HAZ samples were simulated using Gleeble simulator at different welding conditions of heat input and peak temperature. Base steel was prepared through normalizing at 1,000º for 30 min and tempering at 750º for 2 hours, and thus the base steel consisted of tempered martensite and carbides. From TEM analysis, carbides in the base steel were identified as M23C6 carbides along the austenite grain boundaries and the lath boundaries, and MX carbides within the laths. The microstructure of the base steel was changed to martensite and δ-ferrite in the HAZs, i.e., the tempered martensite in the base steel was initially transformed to the austenite and the δ-ferrite during continuous heating up to peak temperature of 1,300º, after which the austenite was transformed to the martensite during rapid cooling. In addition, autotempered martensite and M3C carbides were partially observed in the HAZs. Meanwhile, the volume fraction of δ-ferrite, the prior austenite grain size (PAGS) and lath width of martensite increased with increase in the heat input and peak temperature. The impact toughness of the HAZs deteriorated due to the formation of δ-ferrite and martensite as compared to that of base steel, and it deteriorated further with increase in the heat input and peak temperature. Post Weld Heat Treatment (PWHT) improved the impact toughness of the HAZs; however, its value was still lower than that of base steel due to the remained δ-ferrite.

Id 671

Abstract Final Nr. P3.165

Effect of Ti on Microstructures and Mechanical Properties of 9Cr-1WVTa Reduced Activation Ferritic-Martensitic Steels

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In this study, the effects of Ti, which is one of low activation alloying elements, on microstructures and mechanical properties have been investigated. An attempt has also been made to control carbide precipitation in order to improve mechanical properties, especially charpy impact toughness. All tested steels were designed based on thermodynamic calculations of phase equilibria. Reduced activation ferritic-martensitic steels such as Europer97 and F82H are composed of tempered martensite and various precipitates like relatively coarse M23C6 at grain and lath boundaries and fine MX precipitate within laths. Relatively coarse M23C6 carbides were considered to be mainly harmful to charpy impact properties. Addition of Ti and decrease of carbon content could improve charpy properties significantly, resulting from transformation of coarse M23C6 into fine Ti precipitates.

Id 680

Abstract Final Nr. P3.166

Effect of strain rate on nanoindentation hardness of reduced-activation ferritic steel after ion-irradiation

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In blankets for magnetic fusion device, structural materials may suffer from large electromagnetic force with high strain rates due to plasma disruption evenet. In previous research, we evaluated the dynamic tensile properties of the reduced-activation ferritic steel F82H (submitted to J. Nucl. Mater.) using a high-speed tensile machine and revealed a strain-rate sensitivity of the tensile properties of unirradiated material up to 1400 s-1 at temperatures of 293 K and 423 K. It is concern whether irradiation can change the strain-rate ensitivity or not. In this study, we investigates the strain-rate dependence of nanoindentation hardness of F82H after ion-irradiation with 10.5 MeV Fe3+at 543 K up to 5 dpa (nominal at 1000 nm depth). Nanoindentation hardness tests were carried out at the ambient temperature using Nanoindenter G200 with a continuous stiffness measurement. Strain-rate jump method developed by Maier et al. were also examined to obtain the strain-rate sensitivity of nanoindentation hardness. As a result of strain-rate jump tests with 10 indents on each sample, when the strain-rate changes from 0.004-0.006 to 0.05 s-1 at a depth of 340 nm, the nanoindentation hardness increased approximately by 0.13 GPa for unirradiated F82H and 0.15 GPa for irradiated one. Indeed, the strain-rate sensitivity of F82H was slightly (but meaningfully) increased by the ion-irradiation.

Id 686

Abstract Final Nr. P3.167

Microstructure characteristics of dissimilar friction stir welded joint of ODS ferritic steel and RAF/M steel F82H

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Dissimilar joints of ODS (oxide dispersion strengthened) ferritic steel and RAF/M (reduced activation ferritic-martensitic) steel F82H would be required for the fusion reactor application to increase the flexible design margin and get the benefits from each material in a functional way. Conventional melting welding cannot be used to weld these two materials, because it can disturb the fine dispersion of the oxide particles in the ODS steel, as well as generate brittle intermetallic compounds on the F82H side. FSW (friction stir welding) has been considered to be a promising way to weld these dissimilar steels. So far, no related work about dissimilar FSW of the ODS ferritic steel and F82H steels has been reported. The materials used in this study were 15Cr-ODS ferritic steels and RAF/M steels F82H. The FSW joints were done by different welding parameters in the butting configuration. The welding defects, hardness and Microstructure of dissimilar joints welded by varied parameters were investigated. Results show that the retreating side of the dissimilar joint is the prone area of welding defects. Particularly, wormholes defects tend to emerge in the RS-TMAZ under certain conditions. As the welding temperature of the stir zone is in the range of AC1~AC3 of F82H steel, F82H grains in the stir zone can transform into very fine martensite grains, which possess high hardness, while the HAZ is the lowest area in hardness, as the grain coarsening in this region is remarkable. In all the welding conditions, no intermetallic compound is generated in the joints.

Id 689

Abstract Final Nr. P3.168

Annealing effect on irradiation hardening of 15Cr-ferritic ODS steels under ion-beam irradiation environment

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Oxide dispersion strengthened steels (ODSS) have been developed for application to fusion reactor. In ODSS, the grain size as well as the dispersion morphology of oxide particles greatly influenced their mechanical properties. However, the effect of recrystallization annealing on irradiation performance was not investigated in detail. In this research, the effect of annealing on irradiation hardening of 15Cr-ODSS is investigated. The materials used were Al-free ODSS and Al-added ODSS. Each material is prepared for three type specimens; (1) as-received, (2) annealed and (3) 20% cold rolled after the annealing. The annealing temperature is 1350°C for Al-added ODSS, and 1400°C for Al-free ODSS. Ion-irradiation experiment is performed with DuET facility at Institute of Advanced Energy, Kyoto University. Single-ion beam of 6.4MeV Fe3+ for displacement damage and also a dual-ion beam of 6.4MeV Fe3+ions simultaneously with energy-degraded 1.0 MeV He+ ions were irradiated at 300°C and 470°C. The Al-free ODS steel never recrystallized during the annealing at 1400°C for 1 hr, indicating that the grain size is not significantly changed, while dislocations were annihilated. The hardening by ion-irradiation tended to be decreased by the annealing. In the annealed and then 20% cold rolled specimen (3), the hardening is slightly increased both before and after irradiation in comparison to (2) specimens, which is considered to be due to the increase in the dislocation density caused by the cold rolling. The hardening by helium implantation appears to occur in the dual ion beam irradiated specimens. In Al-added ODSS, Al-Ti-O instead of Y-Ti-O is mainly formed and the dispersed oxide particles of Al-added ODSS are bigger in size and lower in the number density than in Al-free ODSS. This difference in the oxide particles dispersion morphology is reflected in the recrystallization behavior as well as other mechanical properties including tensile strength, hardness and irradiation tolerance.

Id 750

Abstract Final Nr. P3.169

Correlation of microstructure characteristics and mechanical properties of high-Cr ferritic ODS steels and SUS430 stainless steel during 475ºC thermal aging

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High-Cr ferritic steels are of technological interest for structural components of fusion and fission nuclear reactors. It is, however, considered that their high performance in extreme conditions (high temperature, irradiation and corrosion) may be limmited by α/α' phase decomposition. Since oxide particles work as trapping site for point defects, it is considered that the nano-sized particles influence diffusion of Cr atoms. In this research, the phase decomposition behavior in thermally aged high-Cr ferritic steels is investigated with focusing on the effect of oxide particles on the aging embrittlement behavior. Materials used in this research were 12Cr and 15.5Cr oxide dispersion strengthened (ODS) ferritic steels and SUS430 stainless steel of which the Cr concentration is around 16wt.%. Specimens were isothermally aged at 475º up to 10000 hrs. TEM and SEM observations, micro-hardness and tensile tests were carried out before and after aging. Thermal aging caused a significant increase in the hardness and strength in the order of SUS430, 15Cr-ODS, 12Cr-ODS steel. It is noticed that almost quadruple larger hardening is required for the loss of total elongation of 15Cr-ODS than SUS430, indicating high resistance to aging embrittlement in ODS steels. The formation of isolated particles of the chromium-enriched α' phase was observed by TEM. The correlation of phase decomposition and age-hardening in 15Cr-ODS and SUS430 was interpreted by the Orowan type dispersion strengthening model, which indicates that the age-hardening is mainly due to phase decomposition. Service life prediction is vital for ODS steels. Based on a preliminary analysis, nucleation rate determines the aging time necessary for occurrence of embrittlement. Longer time is required for the embrittlement of 15Cr-ODS than SUS430 because of the lower nucleation rate of α' phase in 15Cr-ODS steel.

Id 751

Abstract Final Nr. P3.170

Thermal properties of tungsten- F82H steel joint formed by underwater explosive welding

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Tungsten coating will be applied to reduced-activation ferritic-martensitic(RAFM) steel components of blanket first wall to protect the surface against particle and heat flux. The authors have reported a coating of 0.2 mm thickness tungsten foil on the F82H RAFM steel using underwater explosive welding technique [D. Mori et al, ISFNT-11]. Microstructure of the interface showed a wavy structure typical for this weld technique. This study reports the thermal conductivity and resistance to heat load of this tungsten-F82H steel joint. The laser flash method revelaed that the thermal conductivity at the ambient temperature of the W-coated F82H was 28.6 W/mK, which was slightly larger than that of F82H (27.1 W/mK). Thermal load tests using a YAG laser with a beam spot size of 1.2 mm diameter were carried out on the tungsten side of the rod specimens (3 mm ƒÖ ¡Ñ 3.2 mm height) in vacuum. The back side (F82H side) was put on a water-cooled copper heat sink. After the laser irradiation with 1.7MW/m2 for 30 s, the specimen temperature reached 500 ¢XC measured by a thermocouple contacted with a side surface of F82H. After cooling, the surface and cross-section of the specimen were observed by scanning electron microscopy, resulting in no marked change in the tungsten and the interface of joint. After the laser irradiation with 4MW/m2 for 35 s, while the increase of temperatrue at F82H was smooth until 1000 ¢XC, SEM observation showed that the tungsten coating peeled off from F82H just below the laser spot , possibly due to the siginificant expansion of F82H by the ƒ×-ƒÑ (or ƒÑ¡¦) transformation during cooling. This result suggests that underwater explosive welding is attractive for the fabrication of tungsten coating on RAFM steels for the first wall of blankets.

Id 779

Abstract Final Nr. P3.171

Stress corrosion cracking of structural materials in supercritical water dissolved with hydrogen

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Most of the previous research on stress corrosion cracking (SCC) was conducted in supercritical water (SCW) dissolved with oxygen. However, recent research indicated that non-sensitized SUS316L suffered TGSCC in hot water dissolved with hydrogen in slow strain rate test (SSRT). Since the cooling water of fusion blanket contains tritium, SCC may occur in the structural materials. In this research, therefore, the SCC susceptibility was investigated for the candidate structural materials in SCW dissolved with various hydrogen contents. The materials used are SUS316L austenitic steel, F82H ferritic martensitic steel and SOC-16 oxide dispersion strengthened (ODS) ferritic steel. To evaluate the SCC susceptibility, the SSRTs were carried out in a SCW at 773 K under a pressure of 25 Mpa. The contents of dissolved hydrogen (DH) in SCW were DH $<$ 0.01 ppm (deaerated), DH 0.4 ppm and DH 1.4 ppm. After SSRT, the deformation and fracture behavior analysis were conducted by scanning electron microscope (SEM) and and electron backscatter diffraction (EBSD). In SCW dissolved with hydrogen below 0.01 ppm, the small amount of SCC was observed in SUS316L at a lower strain rate. On the other hand, the fracture mode change was not identified in the ODS steel and the F82H at the tested strain rate range. And more severe oxidation occred on F82H ferritic - martensitic steel than the SUS316L and SOC-16.

Id 896

Abstract Final Nr. P3.172

Investigation of chemical state and distribution of Li in Pb-Li ingots using SXES and rf-GD-OES

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Liquid lead–lithium (Pb-Li) blankets have attracted attention in design concepts such as the Helium-Cooled Lithium Lead (HCLL) in EU, the Dual Coolant Lithium Lead (DCLL) in US, the Dual-Functional Lithium Lead (DFLL) in China, and biomass fusion hybrid concept (GNOME) in our group. There are some issues concerning the compatibility of liquid Pb–Li with other system materials. It is known that Li2O can react with Al and Si (oxides) to form compounds, such as LiAlO2, Li2SiO3, and Li4SiO4. Therefore chemical state of lithium in liquid Pb–Li should be evaluated to certificate the quality of the Pb-Li. As original production of liquid blanket, Pb-Li ingots was studied for their Li distribution and chemical state. In this paper, soft-X-ray spectrometer (SXES) equipped with electron-probe micro-analyzer (EPMA) and radio frequency glow discharge optical emission spectrometry (rf-GD-OES) are utilized to investigate the two-dimensional and depth distribution of Li element and its chemical state in the original Pb-Li ingots. From the measurement of rf-GD-OES Pb-Li ingots have surface layer even if they are kept in glove box (O concentration≤0.1ppm). C,O and Li element have a similar tendency in depth profile. Soft X-ray emission spectrum of pure Li2O,Li2O2 and Li2TiO3 powder was measured by SXES as a reference standard which was compared with that of Pb-Li ingots. Li-satellite peak was detected and identified as 49ev which means the soft X-ray emission from Li-O bond. By using this satellite peak, Li-O bond mapping was successfully obtained. The result showed that in the surface of Pb-Li ingots non-uniform distribution of Li-O bond existed. This result of two-dimensional micro-mapping of Li-O bond obtained as a Li-satellite peak on the surface of Pb-Li suggests possible Li-oxide compounds that may affect corrosion behavior.

Id 469

Abstract Final Nr. P3.173

Tensile properties of F82H steel after aging at 673 to 923 K for 100 kh

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Reduced-activation ferritic/martensitic (RAFM) steels have been developed as the structural materials for fusion blanket. Precipitation during heat treatment and under operation condition affects the strength of RAFM steels. Aging tests have indicated that additional precipitation of MC (M = Ta etc.) hardened the steels, while precipitation of Laves phase (Fe2W) reduced the hardness due to loss of solution hardening by W. However, quantitative analyses for the change in tensile strength have not been conducted yet. The present study seeks tensile properties and deformation mechanisms after the long term aging. 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 1313 K for 2.4 ks and tempering at 1023 K for 1 h. F82H was aged from 673 to 923 K for 100 kh. 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 air, while high temperature tests were perfomed from 673 to 923 K in a vacuum better than 10-4 Pa. The initial strain rate in the tensile tests was 6.7 x 10-4 s-1. Yield strength (YS) and ultimate tensile strength (UTS) at RT before aging was 499 and 648 MPa, respectively. YS at RT after aging at 673, 773, 823, 873 and 923 K was 505, 526, 501, 408 and 309 MPa, respectively. While, UTS at RT after aging was 663, 664, 638, 561 and 483 MPa, respectively. No degradation of tensile strength was observed up to 823 K acording to the YS and UTS, however they were decreased at 873 and 923 K.

Id 895

Abstract Final Nr. P3.174

Hydrogen Isotope Permeation through Structural Materials of a Fusion Reactor

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For fusion devices (such as DEMO and Fusion Neutron Sources) high neutron fluences as well as hydrogen and helium generation in structural materials are expected. New structural materials for the such conditions are required. The tritium is component of fusion fuel available only in small quantities and expensive. Therefore for use of the material in fusion, the parameters of hydrogen permeation should be investigated to predict tritium losses and develop methods of reducing the hydrogen permeation. In the present work gas-driven permeation (GDP) and plasma-driven permeation (PDP) of deuterium through structural materials of three types: reduced activation ferritic-martensitic steel Rusfer-EK-181 (Fe-12Cr-2W-V-Ta-B), austenitic steel ChS-68 (Fe-16Cr-15Ni-2Mo-2Mn-Ti-V-B) and V-4Cr-4Ti alloy were investigated. All materials are developed by A.A. Bochvar’s Institute (Russia). In the experiments RUSFER-EK-181 and ChS-68tubes of 250 mm length with diameter of 6.85 mm and wall thicknessof 0.4 mm (effective area 50 cm2) and V-4Cr-4Ti membrane with diameter of 50 mm and thickness of 0.1 mm (effective area 20 cm2) were used. The gas-driven permeation was measured at temperatures in a range of 600 - 900 K and deuterium pressures in a range of 10-2 - 102 Pa. The distributed ECRH plasma discharge was used for cleaning the inlet surface of samples by argon ions with ion energy 300 eV and for deuterium plasma irradiation in PDP measurements at membrane potential from floating to -300 eV. The plasma-driven permeating flux was measured at temperatures in a range of 600 to 900 K. Deuterium ions current density was $5{\cdot}10^{19}$ ions/m2 at accelerating potential of –300 eV. At GDP permeability coefficient of V-4Cr-4Ti membrane is 4 orders of magnitude higher than of RUSFER-EK-181 while permeability coefficient of ChS-68is higher than permeability coefficient of RUSFER-EK-181 at all pressures and temperatures. The features of such properties are discussed.

Id 721

Abstract Final Nr. P3.175

Laser refrection measurement on liquid lithium flow surface flow

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In the internaitional fusion materials irradiation facility (IFMIF), 14 MeV neutrons are generated by 35-40 MeV deuteron beam injection into a high-speed liquid lithium (Li) plane jet, flowing along a vertical concave wall in vaccum. 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 this study, laser refrection method for a measurement of the surface wave of the Li jet is proposed and tested experimentally. 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. In an experiment, a semiconductor laser with wavelength in 488 nm was used for measurement of surface characterstics of the Li jet. The laser beam was forcused on the Li surface and was reflected. The reflection point was detected as a laser spot on a diffuser plate set at the height of 25 mm from the Li surface and was recorded using a high speed video camera. A slope angele of surface waves was estimated from position data of the reflection point, and then the Li surface fluctuation in a time sequence was calculated using the slope angle and the velocity of the Li jet. It was found that the angle of slope in flow direction is larger than that in transcerse transverse direction . In addition, distributions of wave frequencies and maximum wave heights were calculated with flow velocity. It could be confirmed that these were found to corresponded to the result of those obtained from contact measurement with an electro-contact probe appratus. As a result, proof of principle experiment of laser reflection was performed successfully.

Id 709

Abstract Final Nr. P3.176

Modeling radiation damage in tungsten for fusion plasma facing applications

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Tungsten is the leading solid material for fusion plasma facing component applications because it has many desirable thermo-physical properties compared to other candidate materials. Plasma facing components in fusion power systems must tolerate an extremely hostile environment that includes severe and variable heat loads, damage from neutron bombardment, and surface modifications driven by energetic particles escaping the plasma. A fusion-relevant neutron source does not exist so it is essential to develop predictive models of property evolution as a function of neutron flux and fluence, temperature, and gaseous transmutants in order to interpret experiments using alternative irradiation sources. Molecular dynamics simulations were performed to characterize displacement damage in tungsten induced by 14 MeV neutron bombardments. The effects of primary-knock-on atom (PKA) energy, temperature, and He content on cascade development was studied and compared with three other body-centered cubic metals. We find cascade morphology depends sensitively on PKA energy, and we observe a morphological transition energy that scales with the logarithm of the displacement threshold momentum, defined as $\ln(\sqrt{2E_d m})$. Interestingly, cascade morphologies for tungsten in the high-energy regime were distinctly different compared to the other metals, which has implications for long-time microstructure evolution. To explore cascade-aging, annealing simulations were performed using a specially developed kinetic Monte Carlo code called kSOME. The code is designed to treat the migration, emission, transformation and recombination of point defects and their complexes originating in displacement cascades. Annealing simulations were performed to 10 ns at the same temperature as the cascade simulation. The time dependence of annealing is broadly characterized by early recombination of close proximity anti-defects followed by gradual long-time recombination and migration of self-interstitial atoms out of the simulation box. The detailed annealing behavior depends sensitively on defect migration parameters, and the sizes and spatial distributions of point defect clusters.

Id 851

Abstract Final Nr. P3.177

Beryllium oxidation model implemantation for the gamma-fr code

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The GAMMA-FR code is a Korean domestic system safety analysis code to predict the thermal hydraulic and chemical reaction phenomena expected to occur during the thermo-fluid transients in a nuclear fusion system. A safety analysis of the Korea TBS (Test Blanket System) is underway using this code. GAMMA-FR is a branch of the GAMMA+ code, which is well validated safety analysis code for a HTGR(High Temperature Gas-cooled Reactor). Fusion reactor oriented features includes the capability to address water freezing, air condensation, beryllium, carbon and tungsten oxidation in steam and air environments, flow boiling in the coolant loops, and radiation in enclosures. This paper covers beryllium oxidation model in case of ingress of water/steam into the vacuum vessel. In this accident, hydrogen is formed due to reactions between the water vapour and hot beryllium, carbon and tungsten materials in the vacuum vessel. For situations with presence of air there is a risk of hydrogen explosion.

Id 976

Abstract Final Nr. P3.178

Neutronics Requirements for a DEMO Fusion Power Plant

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The European Power Plant Physics and Technology (PPPT) programme, launched initially by the European Fusion Development Agreement (EFDA) and organised now within the newly established Eurofusion Consortium, aims at developing a conceptual design of a fusion power demonstration plant (DEMO) within the time period of the “Horizon 2020” roadmap. Various integrated PPPT projects are being conducted to meet this ambitious goal including e. g. Breeder Blanket (BB), Safety and Environment (SAE), Magnets (MAG), Materials (MAT), Remote Maintenance (RM), and others. Neutronics plays an important role for all of the related activities since it has to provide essential data which are required for the nuclear design of DEMO and its components, its performance assessment and verification. This paper addresses the neutronic requirements a DEMO fusion power plant needs to fulfil for a reliable and safe operation. The major requirement is to ensure tritium self-sufficiency taking into account the various plant-internal losses that occur during DEMO operation. A further major requirement is to ensure sufficient protection of the superconducting magnets against the radiation penetrating in-vessel components and vessel. To this end, reliable criteria for the radiation loads need to be defined and verified to ensure the reliable operation of the magnets over the lifetime of DEMO. Other issues include radiation induced effects on structural materials such as the accumulated displacement damage, the generation of gases such as helium which may deteriorate the material performance, as well as the production of radioactive nuclides which impact the safety and maintenance. The paper discusses these issues and their impact on design options for DEMO taking into account results obtained in the frame of the 2013 PPPT activities with DEMO models employing the helium cooled pebble bed (HCPB), the helium cooled lithium lead (HCLL), and the water-cooled (WCLL) blanket concepts.

Id 333

Abstract Final Nr. P3.179

Preliminary safety studies for the DEMO HCPB blanket concept

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Helium Cooled Pebble Bed (HCPB) blanket concept is one of the DEMO (Demonstration Power Plant) blanket concepts running for the final design selection. Concept relevant safety needs to be addressed at the early stage of the design. In this paper the preliminary safety studies for the current concept have been performed with respect to the confinement strategy, FMEA (Failure Mode and Effect Analysis), selection of critical event sequences and identification of source terms. Based on ITER, confinement strategy and safety functions have been proposed. Confinement barriers and associated systems are defined to protect the personnel, public and environment against radioactive material releases. FMEA has been done to ensure that a full range of potential faults and off-normal conditions have been considered. The approach is based on a FMEA at component level with evaluation of loss of functions. From FMEA representative accident initiators have been identified and hence Postulated Initiating Events (PIEs) can be listed assessing elementary failures related to different components. Based on these PIEs critical event sequences have been selected. Concerning the event causes, scenarios by failure assumptions, possible consequences and mitigating actions, a priority list of event sequences has been performed for deterministic analyses in the future. Station blackout and bounding accident have been proposed as well; even they are not identified by the current FMEA. They are the most severe conceivable accidents driven by in-plant energy that can lead to the maximum radiological doses. Source terms have been identified as energy, tritium, activation products, dust, ACPs (Activated Corrosion Products) and neutron sputtering products. They are described with respect to the generation, treatment and available inventories - mainly based on the previous studies. The safety is a continuous process accompanying the design development. Open issues that have to be dealt with following the design are pointed out finally.

Id 568

Abstract Final Nr. P3.180

Application of the R2Smesh approach for the accurate estimation of photon radiation dose fields around activated iter components

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The R2Smesh ("Rigorous 2-step") approach, developed by KIT for the determination of shut-down dose rate distributions on high resolution mesh grids, can be utilized to provide photon radiation dose fields around activated components which are moved from the irradiation site in ITER to some external location. R2Smesh couples MCNP transport calculations (neutron and decay gammas) and FISPACT inventory calculations (decay gamma source) through suitable interfaces utilizing MCNP's mesh tally capabilities. Neutron fluxes and the decay gamma sources are provided on high resolution mesh grids superimposed to the real geometry. Thus proper account is taken of the spatial variations of the flux and the decay gamma source distribution without the need to modify the MCNP geometry model. This feature enables to export the decay gamma source distribution from the irradiation site in the reactor to any external location for the determination of photon flux and dose rate distributions. The decay gamma source distribution, provided for a specified irradiation history of the considered ITER component, is then overlaid to the model of the component at the considered location. In this work, we applied the R2Smesh approach to dose rate analyses of various in-vessel components which were activated in ITER and then moved to a transfer cask. The considered components - a divertor cassette, a first wall module, a generic diagnostic port plug and a test blanket module- were assumed to be irradiated in ITER following the standard SA2 irradiation scenario. The activated components were then removed from the ITER torus sector model and integrated into a transfer cask model. The associated decay gamma source distributions were likewise transferred to the cask and overlaid to the geometry models of the activated components using the mentioned feature of the R2S approach. Spatial distributions of the resulting gamma flux and the biological dose rates were generated on detailed high-resolution maps in and around the transfer cask loaded with the different activated components. Related results included in the paper were obtained in the frame of an ITER service contract and need to be validated by the ITER Organisation.

Id 800

Abstract Final Nr. P3.181

Dose rate analysis for the diagnostic generic equatorial port plug in the port plug test facility

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The port plug test facility (PPTF) in the ITER hot cell building will be used to test activated equatorial port plugs (EPP). The EPP will be installed in the test tank of the PPTF in the test tank room. This room is neighboured by the docking room in the front and the service room in the back. Personnel access to these two rooms must be guaranteed when an EPP is installed. Thus the docking room will be designated as “green zone” with limit of the biological dose rate less than 25 μSv/hr. In this work, the R2Smesh approach, developed by KIT for the determination of shut-down dose rate distributions in and around activated components, is used to provide high-resolution maps of the photon flux and dose rate distributions in the test tank room, the docking room and the service room, assuming an activated diagnostic generic equatorial port plug (GEPP) module in the PPTF. The GEPP is assumed to be irradiated in ITER following the standard SA2 irradiation scenario. The R2Smesh system is used to simulate the activation of the GEPP module in ITER thus providing the 3D decay gamma source distribution at the considered cooling time of 30 days after irradiation. The GEPP is then removed from the ITER torus sector model and integrated into the PPTF model. The associated decay gamma source distribution, calculated for the GEPP in ITER, is likewise transferred to the PPTF and overlaid to the GEPP model in the test tank using a unique feature of the R2Smesh approach. Spatial distributions of the resulting gamma flux, the biological and absorbed dose rates are then determined around the activated GEPP module in the test tank room, through the docking and the service rooms. In the paper, detailed high-resolution maps are presented and measures are discussed that may be considered for improving the shielding of the radiation emitted from the activated port plug in the test tank.

Id 800

Abstract Final Nr. P3.182

Evaluation of tritium transport in the biomass-fusion hybrid system and its environmental impact

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The authors have proposed the biomass hybrid fusion system which utilizes the energy generated by fusion reaction for conversion of organic compounds to hydrogen and synthetic fuel by endothermic reaction. This study evaluates the tritium behavior in the plant and its environmental effects and compare with electricity generation. In both systems, major tritium migration path is the permeation through the heat exchanger from the blanket primary coolant to the secondary heat transfer media. In the biomass conversion system, blanket breeder and coolant is LiPb, and secondary medium is liquid lead. Tritium permeation rate was evaluated from the experimental data for SiC composite and metals used for heat exchanger. Permeation rate is also affected by the tritium solubility in LiPb, that is the function of the recovery efficiency of tritium from the breeder stream. Hydrogen as the primary product of biomass conversion is contaminated with tritium, and its concentration should be controlled significantly below the regulation limit, because it goes to the market as consumer product. Finally, tritium is released to the environment from transportation or fuel cells, that will distribute in broad area. In the case of electricity generation, tritium release to the environment in the normal operation is dominated by the heat rejection systems that discard waste heat from condenser of turbines. Released tritium will diffuses from the site of generation station, and affected area expands in the long term operation of the plants. The results suggest the identified design criteria of fusion plants from the environmental tritium aspects for biomass-fusion hybrid and electricity generation. It should be noted that public acceptance of the increased tritium level in the background could be a potential issue for fusion.

Id 690

Abstract Final Nr. P3.183

Parametric Stress Analyses for a Piping of ITER subjected to Seismic Displacements

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ITER (International Thermonuclear Experimental Reactor) to demonstrate commercial energy production from the fusion has been being constructed since 2010. There are many piping in the ITER and some parts of them penetrating adjacent buildings are subjected to large seismic displacements, so that a refinement of layout is one of the design concerns for maintaining safety and reliability of the particular piping. The objective of this study is to determine an optimum layout for a radioactive liquid transfer piping to withstand a given seismic displacements in vertical as well as horizontal directions combined with internal pressure and thermal expansion. To do this, a series of finite element analyses for the piping were performed by changing layouts. In addition, the feasibility for utilizing the double-walled structure was also investigated. Analysis results show that effects of the internal pressure and thermal expansion on total stress were very small compared to that of seismic movements. Also, the stress as well as the deformation of the double-walled piping was larger than that of the single-walled piping although the difference was not big. Therefore, it can be concluded that two configurations such as spiral + U shape and simple elbow + U shape are the most appropriate. Besides, the U-shape can be an alternative option in installation and maintenance of the piping when considering the simplicity.

Id 230

Abstract Final Nr. P3.184

Status of Safety Studies and Code Developments towards Korean Fusion DEMO Plant

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South Korea aims at the construction of K-DEMO using the experience of the project of KSTAR and ITER. The findings of above mentioned projects would be applied for the the verification of component and economic concerns. In this paper, the status of safety researches performed by National Lab-University cooperation is summarized into two parts. First, we are performing the conceptual design of safety systems for K-DEMO through tracking ITER licensing process and establishing the direction of future safety studies. General information of regulation and licensing for ITER was identified through the preliminary safety report (in French, Rapport Preliminaire de Surete, RPrS) and the safety issues for procurement/non-procurement items has been considered, reviewed and updated by Korean fusion safety advisory group. Based on this informaiton, we have been performing safety studies for K-DEMO using Phenomena Identification and Rankign Table (PIRT) and Ovjective Provision Tree (OPT) that is one of tools of Integrated Safety Assessment Methodology (ISAM) proposed by Generation IV Forum Risk and Safety Working Group (RSWG). Second, safety analysis codes and models for K-DEMO were being developed. Curenttly, KAERI (Korea Atomic Energy Research Institute) is developing GAMMA-FR based on GAMMA for fission reactor analysis. KAERI has requested MELCOR-Fusion 1.8.2 to U.S. NRC for V&V of GAMMA-FR. While the purpose of GAMMA-FR development is to secure design basis, universities have a role to perform the corss-validation using MELCOR-Fusion and GAMMA-FR, and develop the initial setup of safety analysis models for K-DEMO. Finally, the long-term project of technology development for K-DEMO is being planned for 10 years. In this planning, 'safety and license division' is planning R&D program for quality assurence, licensing process, maintenance and repaire including above contents.

Id 921

Abstract Final Nr. P3.185

Safety studies on Korean fusion DEMO plant using integrated safety assessment methodology: Part 2

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The purpose of this paper is to suggest methodology that can investigate safety issues and provides a case study for Korean fusion DEMO plant (K-DEMO). A basic step of the regulation and licensing procedure for demonstration of nuclear facilities is a solid establishment of safety philosophies. Even though nuclear regulation and licensing framework is well setup due to the operating and design experience of Pressurized Water Reactors (PWRs) since 1970s, the regulatory authority of South Korea has concerned on the challenge of facing new nuclear facilities including K-DEMO due to the differences in systems, materials, and inherent safety feature from conventional PWRs. In licensing process for safety philosophies, the regulator should identify the gaps between ITER and DEMO in terms of safety issues. For that reason, we performed to track ITER licensing process and establish the direction of future safety studies. A general information of regulation and licensing for ITER was identified through the preliminary safety report (in French, Rapport Preliminaire de Surete, RPrS) and safety issues for procurement/non-procurement items has been considered, reviewed and updated by Korean fusion advisory group with specialists from domestic universities, industries, and national institutes. Based on this informaiton, we have performed safety studies for K-DEMO using Phenomena Identification and Rankign Table (PIRT) and Ovjective Provision Tree (OPT) that is one of tool of Integrated Safety Assessment Methodology (ISAM) poposed by Generation IV Forum Risk and Safety Working Group (RSWG). Considering the design phase of K-DEMO, the current study focused on the establishment of safety philosophy, PIRT and OPT for K-DEMO with the advisory group in South Korea.

Id 922

Abstract Final Nr. P3.186

Argon generation in fusion reactor materials

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Different candidate plasma facing materials (as tungsten, beryllium), the low activation structure materials (as vanadium alloys, silicon carbides), liquid breeders (lithium and lithium-lead) and some others have been suggested for future fusion power reactor cores as corresponding to maintenance, recycling and for waste disposal acceptance after 50 and 100 years of cooling. It is shown by the neutron activation analysis that a relatively short-lived Ar-41 (T1/2=1.85 hour) and rather long lived Ar-42 (T1/2=33 yr) and Ar-39 (T1/2=269 yr) may appear in these materials under the fusion neutron irradiation conditions. While argon production is essentially less than helium production in irradiated materials, at other times its role, e.g., in the inhalation dose, becomes significant. In some cases the Ar-39 activities is comparable or even exceeds the C-14 activity and may become apparent after tritium removal from plasma exhaust and dust, from the liquid breeders, during plasma-facing and structural component recycling and waste management. The main source terms of argon-39 activity for these materials were identified and the specific production rates were evaluated. The study shows that the specific Ar-39 activity induced in the materials is significantly dependent upon the assumptions for unavoidable impurity contents or constituents as potash and calcium. Therefore the initial impurity control is definitely recommended. Otherwise a long-term storage for the rare argon radionuclides considered essential.

Id 855

Abstract Final Nr. P3.187

Development and application of the activation and decay database for corrosion products simulation of fusion reactor

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Studies show that activated corrosion products (ACPs) are the main source of radiation during the normal operation and maintenance, as well as accidents of fusion reactors. For the source term analysis of ACPs in the cooling loop, a specialized activation and decay database is required, which is directly related to the material composition and neutron energy of fusion reactors. Therefore, in this paper, the types of the potential materials used in the cooling loop of fusion reactors were collected firstly, including those used at present (304 SS, 316L SS, Inconel 690, CuCrZr alloy, etc.) and also used in the future (low activation steels, vanadium alloy, etc.). Then the chemical composition of the materials was investigated detailedly and the total number of the chemical elements is 21, such as C, N, O, Cr, Mn, Fe, Co, Ni. Subsequently, the activation and decay reactions of each chemical element occurring under the radiation of neutrons with the energy of 14.1 MeV were tabulated to form the database. Finally, the database was applied to the code named CATE, which was an independently-developed code for the source term analysis of ACPs, for the ITER simulation. The results show that compared to the traditional materials, the low activation materials used in the field with neutron radiation can significantly reduce the radioactivity of ACPs in the cooling loop, and the radioactivity of ACPs from different types of low activation materials are obvious different.

Id 432

Abstract Final Nr. P3.188

Design Basis Accident Analysis for the Ignitor Experiment

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Ignitor is a compact high-magnetic toroidal field experiment aimed at studying plasma burning conditions up to ignition in Deuterium-Tritium plasmas. A safety analysis study has been applied to the Ignitor machine using Probabilistic Safety Assessment. The main initiating events have been identified, and accident sequences have been studied. A deterministic assessment of the main accidental sequences has been performed. The consequences of the radioactive environmental releases have been assessed by means of a population dose code. The purpose of this paper is to analyze the deterministic consequences of some of these accidental sequences, serving as the 'design basis accidents' because of the extent of radioactive release involved, either outside or inside the building. The two sequences with higher releases have been considered. Regarding the outside release (Erroneous opening of the valve of the high pressure tank: stack release) , the value of the individual EDE to the critical group , estimated at 20 µSv , committed in 70 years (due to an accident frequency of 8.5 10-3/y) is negligible. The cumulative dose, committed in 70 years, is equal to 2.3 10-2 Sv-person, negligible too: applying the ICRP risk coefficients we get to estimate as “zero” the number of cancers expected in 70 years in the entire population around the site. Regarding the inside release (Erroneous opening of the valve of the high pressure tank: glove box release) the Acute Dose to the personnel is around 60 µSv, then again the risk to exposed workers, is negligible. The deterministic analysis has achieved the following results: the IGNITOR machine, both during Routine functioning and Accidental sequences, presents a negligible environmental impact and radiological risk. The Ignitor plant does not need further containment building, or any emergency procedure such evacuation or sheltering or food consumption prohibition in the case of an accident.

Id 622

Abstract Final Nr. P4.002

Energy balance of He3 lunar mining for fuelling advanced-plasma fusion reactors

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The recent stress on safety by the world’s community has stimulated the research on other fuel cycles than the Deuterium-Tritium (DT) one, based on ‘advanced’ reactions, such as Deuterium-Helium-3 (DHe3). With these cycles, it is not necessary to breed and fuel tritium. The DHe3 cycle has also a very low presence of fusion neutrons. The plasma confinement requirements for a DHe3 reactor are more challenging than those for a DT reactor, and He3 is not currently available on earth. As an important step towards the study of a DHe3 reactor, a feasibility study of a high-field DHe3 experiment of relatively large dimensions and high fusion power, however based on present technologies, has brought to the proposal of the Candor fusion experiment. Terrestrial sources of He3 are very small. However, a major deposit of He3, originally discovered in lunar samples in 1970 was considered by the fusion program in 1986. The He3 of the moon is contained in the regolith, roughly a million metric tonnes. The approach to the extraction of lunar He3 must be done in such a way that uses properly the available resources, avoids unnecessary energy losses and maximizes the amount of extracted resources. The DHe3 reaction releases 6 105 GJ per fusion, that is 10 MWe year per kg of He3. 40 tons of liquified He3 brought from the Moon to the Earth – about the amount that would comfortably fit in two space shuttles – would provide sufficient fuel for He3 based fusion reactors to meet 25% of the world electrical needs for one year. A cost and energy balance assessment to feed the Candor experiment has been performed, based upon available data. It turns out that He3 feed for a large experiment such as Candor can be a good demonstration technological step for lunar DHe3 mining.

Id 622

Abstract Final Nr. P4.003

Method to measure the composition of H-D-T mixtures employing capacitive sensors

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The hydrogen isotopes separation systems based on cryogenic distillation is one of the key systems within the fuel cycle of a fusion reactor. In the ITER project, the isotope separation system (ISS) is also primary based on cryogenic distillation and supported by catalytic chemical equilibration aiming to separate elemental hydrogen isotopes gas mixtures. Feed streams from various ITER fuel cycles shall be processed within the cryogenic distillation cascade and specific tritium, deuterium and hydrogen based streams shall be delivered at certain chemical specification. For continuous determination of the composition of the hydrogen isotopes mixtures in liquid phase, by measuring the electrical parameters of the mixtures, employing of a capacitive sensor is proposed. The proposed capacitive sensor comprises a measuring sensor and an electronic module. The measuring sensor is of original construction, a 3D matrix capacitor that presents clear benefits on the distribution of the electric field inside it. The measuring electronic module is a flexible option which provides high accuracy of the composition at different temperatures of the mixtures. The paper aims to present the constructive and functional features of the measuring sensor and of the electronic module. In addition, experimental results covering the characteristic composition - permittivity of various concentrations of hydrogen – deuterium will also be provided. Key words: capacitive sensor, cryogenic distillation, hydrogen-deuterium liquid mixture, electric permittivity.

Id 512

Abstract Final Nr. P4.004

Plasma Start-up Design and First Ohmic Experiment in VEST

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Ohmic plasma start-up experiments in Versatile Experiment Spherical Torus (VEST) have been successfully carried out with conventional start-up scheme using central solenoid. VEST has a toroidally continuous vessel as ITER, in which significant eddy current is induced to affect the magnetic field inside the vacuum vessel and limit the loop voltage. Start-up scenarios for conventional start-up in VEST have been developed using a discharge design code that calculate the vacuum field structure considering eddy current induced in the vacuum vessel wall with given power supply parameters or measured PF coil current. The start-up scenarios generate filed null region at the inboard side in the onset of loop voltage in order to maximize the connection length, and provide the required vertical magnetic field for force balance as the plasma current evolves. Hydrogen is used as a working gas and the toroidal field of up to 0.12 T with 0.2 s flat-top is applied on magnetic axis during a discharge. With assist of ECH (Electron Cyclotron Heating) preionization (6 kW, 2.45 GHz), a plasma current of ~60 kA has been generated with volt-second of 23 mV·s. The elongation and edge safety factor are estimated to be 1.6 and 3.7 respectively by equilibrium reconstruction.

Id 844

Abstract Final Nr. P4.005

Conceptual study on design parameters of fusion DEMO based on spherical torus

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Spherical torus (ST) plasma have a great potential of a high beta operation with a compact-sized device. Conceptual study is performed to provide the optimal deisgn parameters for the ST demo and to provide the basis for the ST power plant development path. A self-consistent system analysis coupled with a one-dimensional radiation transport code is utilized as an analyzing tool with limited extrapolations in plasma physics and engineering performance compared to the design basis of ITER. Reference fusion power for the ST demo is set to around 1500MW. Optimum radial build for the minimumm major radius and recirculating power are found by varying aspect ratio, normalized beta value, and confinement enhancement factor. Characteristics of the ST demo design paremteres are compared with cases with low fusion power(~150MW) and commercial fusion power(~3000MW), and viability of a compact-sized STdemo is addressed.

Id 918

Abstract Final Nr. P4.006

Initial operation of the pulsed electron source for helicity injection in Versatile Experiment Spherical Torus at Seoul National University

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Helicity injection is one of the promising start-up method for Spherical Torus (ST) device. In order to study helicity injection in ST, developing a high current electron source is essential. In this study, a high current electron gun using washer stacks has been developed for the helicity injection system in Versatile Experiment Spherical Torus (VEST) at Seoul National University. The developed electron gun can extract electron currents of up to 2 kA with the pulse duration of 10ms. By changing its geometry and operation conditions such as operating pressure and bias voltage, maximum beam currents are obtained. For the successful operation as helicity injector in the VEST device, magnetic field structure is also optimized. The size of the electron source has been minimized (radius of 60 mm and length 80mm) for simplicity and easy installation. Initial operation results of the electron source show the developed electron gun is proper as a helicity injector.

Id 924

Abstract Final Nr. P4.007

Feasibility Study for the Heat Flux Control of ELM-like Plasma Jet Generated by a Pulsed Plasmsa Gun with Transient Electric Field

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Mitigation of severe heat load on divertors during edge localized mode (ELM) events is a critical issue in fusion engineering. In this study, we have investigated the feasibility of active control of ELM-like plasma jets using transient electric field. The ELM-like plasma jet is generated by a parallel-plate pulsed plasma gun, and its electron density and temperature are measured to be ~10^19 m-3 and ~3 eV respectively, slightly depending on the operating conditions. The flow velocity and axial elongation of the plasma jet are also measured to be ~60 km/s and ~80 cm, respectively. Active control of the plasma jet has been performed by applying time-varying transverse electric field so as to induce wiggling motion of the jet by EXB drift. The deformation of plasma jet is observed with diagnostics such as electrostatic probes and photodiode array. Based on the present experimental results, operating parameters for the control of actual ELM plasma are proposed through two-dimensional particle-in-cell (PIC) simulations.

Id 442

Abstract Final Nr. P4.008

Reserch on Localization and Allignment Technology for Transfer Cask

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During plasma discharge in Tokmak Building (TB) of International Thermonuclear Experimental Reactor (ITER), internal components may be suffered from contamination of gamma-ray and tritium. The transfer cask carring pallet and alignment pins is used to navigate between TB and HCB (Hot Cell Building), and to align with the windows of TB to replace components. According to the characters of long length of transfer cask compared to the environment space between TB and HCB, this paper proposed an autonomous navigation and alignment method for internal components transportation and replacement. A localization and path planning method based on localizability estimation is used to realize the cask's localization and navigation accurately. The dynamic localizability matrix is proposed for on-line estimation of localization uncertainty, based on the observation of laser range finder which fixed on the cask and the environment map which built off-line. This matrix describes both the localizability index and localizability direction of cask quantitatively. Based on the localizability estimation, the feasible path with high localizability could be planned out, and guide cask navigate and arrive at destinations through trajectory following. Once cask arrives at the front of the TB window, the position and attitude measurement system is used to detect the relative align error between the seal door of pallet and the window of TB in real-time. According to this offset, the alignment of seal door and TB window could be realized. The simulation experiment based on the real model was designed according to the real TB situation. The experiment results showed that the localization error could be limited in 25mm in xy-direction and 1.5° in heading angle, the alignment error could be limited in 0.9mm in xyz-direction and 0.4° in heading angle, and the proposed system could be used for transfer cask of ITER.

Id 453

Abstract Final Nr. P4.009

Structural Analysis on the ITER Gas Distribution System Manifolds

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The paper presents the thermal analysis and seismic analysis on Gas Distribution System (GDS) manifolds structure which is dedicated pipes combination structure. GDS manifolds for ITER tokamak, which contain individual pipes for injecting gases, evacuation and envelope and supports, distribute gas to Gas Value Boxes from the tritium plant. As required by Safety Requirement Roombook (ITER_D_KF63PB), deadweight, thermal and seismic are essential loads which shall be taken into account. The manifold models for typical sections and whole layout are built for thermal analysis and seismic analysis respectively. Limit load analysis, elastic analysis and elasto-plastic analysis methods are used for safety assessment in accordance with the rules of design-by-analysis. By using the spectrum analysis method, structural stiffness and strength under seismic load have been evaluated. The analysis results demonstrated: (I) manifolds are in plastic stability state under thermal load, (II) the repair or replacement of the straight sections and the sections with T-unit is necessary after Loss of Coolant Accident event except for the elbow sections, (III) manifolds are safe compared with ASME B31.3 and ITER guideline “Allowable values/limits in service level C and D (ITER_D_3G3SYJ)”, (IV) some optimizations to the design are suggested, especially to reduce the thermal stresses in straight section by adding elbows. Keywords: Gas Distribution System manifolds, thermal analysis, design-by-analysis, seismic analysis

Id 59

Abstract Final Nr. P4.010

Assembly study for HL-2M Tokamak

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HL-2M is a new tokamak under construction in Southwestern Institute of Physics (SWIP), the plasma major radius and minor radius is 1.78 m and 0.65m respectively, and the plasma current will reach up to 2.5MA. The toroidal field coils (TFCs) are designed with demountable structure, so the vacuum vessel (VV) and poloidal field coils (PFCs) can be installed integrally. The TFC system comprises 20 D-shaped coils, each coil consists of 7 copper turns. Their wedge shaped inner sections are bonded together to form an integral centre-post which are over-wrapped by center solenoid coil. 20 bundles outer sections are mechanically joined to the center-post. 16 PFCs coils are arranged in the space between the TFCs and VV. D-shaped cross-section VV torus consisted of 20 sectors with double wall structure. The support structure includes a gravity platform, an anti-torque structure and a PFC support frame. The height of HL-2M tokamak is about 8.4m form ground. The 70 tonnes center-post will be transported form its manufacture platform at another hall and located it to its final device gravity platform, 16 tonnes VV body will be inserted from top of TFC center-post to its position. Because maximum crane lift ability is only 20 tonnes, special methods should be used for big parts installation, and special assembly tools are designed for precise assembly of components such as TFC upper sections with finger joints and outer sections with strictly electric contact requirements. The survey and alignment, conversion of survey benchmark, treatment for electric insulation etc have been considered. Assemblies scenarios are studied, virtual assembly analysis have been done. Each subsystems are under developing, the civil construction for device foundation have finished. The installation company is under investigating and will be chosen soon. With the delivery of machine components, the assembly will be carried out.

Id 260

Abstract Final Nr. P4.011

The component development status of HL-2M tokamak

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HL-2M, as a new tokamak of the HL-2A modification and the 2nd step of HL-2 project, is a medium-sized copper-conductor machine under construction at the Southwestern Institute of Physics (SWIP). HL-2M project was approved in 2010. The project includes development of all tokamak components, e.g. toroidal field coils (TFC), poloidal field coils (PFC), vacuum vessel (VV), support, and in-vessel components, etc. Hence it covers development or upgrades of all subsystems for machine operation, e.g. power capability from grid, power station, power supplies, mator-generator (MG), water-cooling, baking, control, cryogenic, etc. HL-2M and HL-2A share on hall with 25 m distance to each other, the projects also implies civil constructions of the machine base underground and shielding. Three sets of diagnostics, a 2 MW ECH system and an office building are also involved into the project. Now the project is progressing. The civil constructions outside the machine hall have been finished. The civil constructions inside, i.e. tokamak base, MG building, and shielding of machine hall, will be finished in Apr. 2014. The MG is ready for delivery and assembly. All tokamak components, e.g. coils, VV and support are being manufactured in industry, with lots of interactions among manufacturers, material providers and SWIP. Many samples, prototypes and tests have been made in advance. The power supplies are being manufactured, delivered and assembled one by one. Other systems, e.g. diagnostics, heating, control, first wall, divertor, water-cooling, baking, cryogenic, pumping, etc. are ready for final design.

Id 437

Abstract Final Nr. P4.012

Progress towards Compact Fusion Energy

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The Spherical Tokamak (ST) is known to be efficient, containing maximum plasma pressure for a given magnetic field, with virtues including improved stability, high elongation and high bootstrap fraction; but to provide significant fusion power a strong magnetic field is required. This necessitates large currents with high resistive loss in the narrow centre column, leading (to date) to very large power plant designs. However several recent developments have led to a new concept: the development of a Fusion Power Plant consisting of several compact ST modules, where in each module the fusion power is small (so that wall and divertor loads are modest) but the gain Q is large. Tokamak Energy Ltd is pursuing the opportunities and challenges of this modular concept. The recent development of high temperature superconductors, when operated at ~20K can provide high current density under strong fields. An HTS magnet requires shielding to limit nuclear heating and damage, however its compactness permits space for shielding, and research into the optimum shield material and configuration will be presented. Recent studies using a Monte-Carlo code with direct integration of gyro-orbital motion show that in an ST rather small plasma currents of 4MA or less may be sufficient to contain the fusion alphas, lower than expected. High energy confinement is key to the attainment of high Q in a small device and results from MAST and NSTX, together with theoretical studies, suggest improvements in confinement under the combination of ST geometry and high magnetic field. Present STs operate at ~0.5Tesla; the MAST and NSTX Upgrades will reach 1T. Tokamak Energy is planning construction of a 3T compact ST to investigate these optimistic predictions. Details will be presented of the novel design features of this first high-field ST, together with progress on world's first all-HTS tokamak ST25(HTS) now being commissioned.

Id 592

Abstract Final Nr. P4.013

Neutronics in support of the bioshield plug design of Equatorial Port 12 for ITER

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In ITER the bioshield is the last radiation barrier before the Port Cell (PC) area, consequently, its design has a large impact in the Shutdown Dose Rates (SDRs) in the PC, which has a limit for human access of 10 μSv/h after 1 day of machine shutdown. Neutronics is the main design driver for the bioshield plug, therefore, challenging neutronic calculations were done taken as a reference model for the study the Diagnostic Equatorial Port 12, which underwent substantial design and integration activities to reach a good level of maturity. The neutron attenuation from the plasma to the PC area is about 9 orders of magnitude which makes computationally very expensive the simulations. In order to run parametric analysis in support of the bioshield plug optimization a two phase approach was taken, doing first the neutron transport from the plasma up to the bioshield in order to get the information needed for creating a new neutron source hence decoupling the PC with the rest of the machine. The second step starts the transport from this neutron source through the bioshield and PC. For the calculation of the SDR the R2SUNED system, based on the rigorous two step (R2S) method with mesh approach , was used. Combinations of different neutron and gamma shielding materials were studied as well as the effectiveness of neutron labyrinths inside the bioshield plugs. The importance of the activation of additional shielding or structural materials in the PC was assessed, as this has additional requirements on impurity levels, affecting the cost.

Id 339

Abstract Final Nr. P4.014

RELAP5-3D©pre-test analysis in support of HE-FUS3 experimental campaign

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The present paper deals with the design of an experimental campaign that will be carried out in He-Fus3 (European helium cooled blanket integral test) facility located at ENEA Brasimone Research Center, in support of the conceptual and preliminary design of the European Test Blanket System (EU-TBS). The results of the experimental campaign are intended to extend the database for the TH-SYS codes validation on the complex thermal-hydraulic phenomena associated with helium loop conditions. In order to establish the test matrix, a numerical activity has been performed using RELAP5-3D© system code. For this purpose, a RELAP5-3D © computational model has been developed and validated on the basis of available experimental results. The pre-test activity focuses on the assessment of the model in normal operating conditions, and allows to investigate the system safety issues in case of accidental scenarios relevant for a TBS of HCS (Helium Cooling System), namely a LOCA (Loss of Coolant Accident) and a LOFA (Loss of Flow Accident) scenarios occurring at different loop operative conditions. The outcomes from the numerical simulations are: 1) the improvement of the facility layout and the optimization of its performances; 2) the design of the experiments having range of TH parameters ITER relevant and simulating accident sequences, where TH phenomena are representative of HCLL system; 3) the enhancement of the data measurement system to detect, to measure and to analyze the expected TH phenomena.

Id 985

Abstract Final Nr. P4.016

Detailed 3-D Nuclear Analysis of ITER Blanket Modules

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In ITER, the blanket modules (BM) are arranged around the plasma to provide heat removal as well as to provide nuclear shielding for the vacuum vessel (VV), magnets, and other components. The BM main components are the First Wall (FW) panel and Shield Block (SB). The design process for the BM includes assessment of thermal stress, detailed computational fluid dynamics (CFD), and electromagnetic (EM) analyses. Re-welding is required at several locations in the BM and the VV behind it and this requires accurate determination of helium production in the structural material. Therefore, detailed calculations of nuclear heating, radiation damage, and helium production are essential for the design process. Following the BM final design review, several areas were identified as needing additional 3-D analysis to be certain nuclear design limits are met. In this work we will use the CAD based DAG-MCNP5 transport code to analyze detailed models inserted into a 40 degree partially homogenized ITER model. The CAD models used are the most up to date models available from the ITER organization or respective domestic agency. Nuclear analysis results will be presented for the NB region (BM14,BM15), the upper ELM coil region (BM11-BM13), the BM03/BM04 waveguide region, and BM18. The results show that the nuclear radiation limits are exceeded in some cases. The dpa in the port extension/VV near the NB region reaches values up to 0.58 dpa. The nuclear heating in the VV behind the upper ELM coil exceeds the 0.6 W/cm3 limit in several regions. He production in some of the BM18 FW/SB coolant connecting tubes exceeds the 3 appm limit at some locations. These results are being used by ITER engineers to perform further analysis of the specified limits and if necessary, to modify the design of BMs or surrounding components.

Id 635

Abstract Final Nr.   P4.017

Mechanical and thermal considerations for the jet li-beam ion source upgrade

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Beam Emission Spectroscopy(BES) is an important method of fusion diagnostic technology. One version of it is Lithium BES widely used for edge plasma density profile and turbulence measurements. Time resolution of the diagnostic is limited by the beam current, therefore new developments are necessary to improve temporal resolution. In the last year, 2013, an upgrade of the JET Lithium-Beam injector has taken place in a collaboration between the Wigner RCP of the Hungarian Academy of Sciences and the Culham Centre for Fusion Energy in the United Kingdom. The element of the upgrade was to replace the ion source with a new type in order to provide longer lifecycle, more extracted ion current with appropiate focusing. A new type of ion source was developed at Wigner RCP capable of providing more extracted ion current (up to 4-5mA) compared to the previously used source which could emit about 2.5mA. The total extractable charge also increased, by a factor of at least a few. This development was made possible by a novel source heater, which could reliably raise the source temperature up to 1370-1380. The increased temperature resulted in higher heating power at a different heating current, changing the operational parameters of the beam significantly. This in turn neccesitated a new cooling system, an additional transformer and other changes which are non-trivial in the JET environment.

Id 716

Abstract Final Nr. P4.018

Sudy of a new concept of ICRH antenna by modelling and experiments

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The proposed new ICRF antenna concept in [1] has been confirmed. In the ANTITER II code a layer of low loss dielectric replaces the artificial dielectric covering the back-plate of the antenna box. The W7-X antenna is taken as a reference example. High Q resonances are obtained for large values of the dielectric constant KD. They correspond to TEn,0 cavity modes of the antenna box, partially filled with dielectric. These resonances can occur when the wave of the considered mode is propagating into the dielectric and evanescent in the vacuum part. This imposes to the operating frequency to be between both waveguide cutoff frequencies. The resonance condition requires the equality of the surface impedance Ey/Hz at both sides of the dielectric-vacuum boundary. The conclusions of the modeling have been experimentally verified on an antenna box partially filled with dielectric, excited either by a strap or a side loop and loaded by a salted water dummy load. The high Q resonances are damped by the dummy load at the aperture of the antenna box but also by the losses in the dielectric. The loading also shifts their resonance frequencies. Therefore: (i) resonance tracking is mandatory either by acting on the dielectric (or equivalent one) or on the generator frequency, (ii) very low loss dielectric or equivalent artificial one is needed to avoid the dissipation of a significant part of the power in it. The strap is not needed to excite the resonance: appropriate side loops can be used and designed to be close to matching for the average loading condition. [1] D. Milanesio and R. Maggiora, Proc. of the 20th RF topical conference (Sorrento, June 2013)

Id 338

Abstract Final Nr. P4.019

Tridimensional Modeling & Numerical Optimization of the W7-X ICRH Antenna

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Ion Cyclotron Resonance Heating (ICRH) is a promising heating and wall conditioning method considered for the W7-X stellarator and a dedicated ICRH antenna has been designed. This antenna must perform several tasks in a long term physics program: fast particles generation, heating at high densities, current drive and ICRH physics studies. Various minority heating scenarios are considered and two frequency bands will be used. In the present work a design for the low frequency range (25-38 MHz) only is developed. The antenna is made of 2 straps with tap feeds and tuning capacitors with DC capacitance in the range 15-200 pF. These capacitors introduce additionnal constraints on the optimization and on the maximum amount of power that can be coupled to the plasma: not only the capacitor voltages cannot exceed a certain value (42 kV) but also the currents are limited to approximately 800 A rms to ensure sufficient heat dissipation for the considered duty cycle. Starting from an initial geometry we used the tridimensional electromagnetic software CST MicroWave Studio (MWS) to assess and optimize its coupling properties. By modifying some geometrical parameters of the front face (strap width, antenna box depth, strap length, strap feeders shape), we show that a substantial increase in maximum coupled power can be obtained accounting for the technical constraints on the capacitors. The various steps of the optimization are validated with TOPICA simulations. For a given density profile the RF power coupling expectancy can be precisely computed.

Id 340

Abstract Final Nr. P4.020

Validation of the electrical design of the W7-X ICRF antenna on a reduced-scale mock-up

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The W7-X ICRF antenna foreseen for operation in the 25-38MHz frequency band is a two strap antenna allowing different toroidal phasings. A very effective prematching is obtained by making resonant strap circuits using external vacuum capacitors. A tap feed leads to a significant decrease of the maximum voltage in the feeding lines. The radiated power is computed from the antenna 4x4 scattering matrix. Its maximum for each considered strap phasing is limited by the maximum current and voltage ratings of the tuning capacitors. The dipole phasing provides the best heating efficiency but has the worst coupling to the plasma or dummy load. A better coupling is obtained by the (0 pi/2) phasing with current drive effect and some edge modes excitation. The monopole phasing with predominant edge mode excitation can be used for wall conditioning. A scaled mock-up (1/4) has been constructed and placed in front of a dielectric dummy load. It allows comparing measured and predicted coupling performances and hence validating the electrical design of the antenna. It gives confidence in the initial simulation results leading to further numerical optimization [1]. High dielectric constant materials are needed for the dummy load to mimic the plasma. Salted water and a mix of the ferroelectric $BaTiO_3$ and salted water [2] are used. The measurements have been compared with the expectations of 3 codes: ANTITER II, MWS and TOPICA. The best agreement is obtained with the $BaTiO_3$ as dummy load for all phasings. [1] F. Louche at al., this conference. [2] H.Bottollier-Curtet et al., Fusion Eng. Des. 86(2011)2651-2654.

Id 642

Abstract Final Nr. P4.021

Compact Toroid Challenge experimental device at the P.N. Lebedev Physical Institute

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Review of compact torus (CT) experiments at the Lebedev Physical Institute is presented. The new method of CT formation and plasma heating is proposed. Recent CTC (Compact Toroid Challenge) experimental results and enhanced CT formation are described, schemes of formation chamber, B-probes and electrical scheme are shown. Such a system could be used as a neutron source, magnetized target in magnetized target fusion, or fusion propulsion system. Typical oscillograms - current in windings, voltage on exploding wire, magnetic fields are shown. The poloidal magnetic field Bp signal; the calculated poloidal magnetic field Bp; the poloidal magnetic field Bp evaluation under main solenoid current It termination; and the simulated value of trapped magnetic field are analyzed. Evolution of magnetic field on the axis of the chamber Bp, poloidal current Ip, the current in the main solenoid It, and filtered poloidal magnetic field Bp are presented. Formation of a compact toroid or field reversed configuration with a maximum energy input into plasma is an important scientific and technical challenge.

Id 78

Abstract Final Nr. P4.022

Method of measurement of neutral beam profile in the long-pulse powerful injectors

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Measurement of neutral beam profiles in powerful long-pulse injectors intended for plasma heating and current drive in thermonuclear installations is a problem task in consequence of very high beam power density (PD) at a level of tens MW/m2. Knowledge of the beam profiles is required for determination of such important parameters as the beam divergence angle and accuracy of its axis aiming onto a tokamak entrance window. However, with so high PD use of probes introduced directly in the beam is impossible. In the ITER heating injector V-shaped calorimeter is designed for the beam commissioning and determination of its parameters. It consists of two panels, each is formed by a set of horizontally oriented cooled tubes arranged in two layers (“front” and “back”). Use of the thermocouples, placed on the tube ends, allows to measure only horizontally averaged NB vertical profile but horizontal profile remains unknown. Given report describes method of measurement of the beam detailed profiles on the calorimeter using matrix of collectors, each of them is placed between tubes of “back” layer in shadow of “front” layer tube, i.e. does not see the beam. Each collector is to collect secondary-emission electrons resulting on surfaces of neighboring tubes under beam bombardment. Such collector matrix enables determination of horizontal and vertical profiles in any section of the beam footprint. Proposed method was tested on the injector stand IREK with tubular V-shaped calorimeter. Measurements were carried out with use of positive hydrogen ion source with beam energy and current at a level of 40 keV/40 À. The calorimeter was equipped by vertical sets of tungsten probes, directly entered into the beam, and secondary-emission electron collectors. Experimental results demonstrated good coincidence of normalized profiles measured by both methods and absence of the profile distortions due to secondary plasma electrons.

Id 643

Abstract Final Nr. P4.023

Long Pulse Performance and Plan of Neutral Beam Injector in KSTAR

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Neutral beam injector (NBI) is playing an important role in supporting the physics studies and long pulse discharge as well as in the beam-based diagnostics such as CES and MSE using beam modulation in KSTAR tokamak. The first NBI (NBI-1) is designed to support the long pulse KSTAR operation with three co-tangential neutral beams with each neutral D0 beam power of 2 MW at 100 keV in the horizontal midplane of KSTAR tokamak with haracteristics of nearly on-axis current drive. The long pulse design requirement requires the large beam sources and efficient water-cooling beamline components collimating beam to allow three beams in effective aperture in the port duct. Presently, NBI-1 has been operated with two beam sources at maximum 2.4 MW beam power with duration of 20 s (3.5 MW for short pulse). The NBI-1 system is under upgrade for the last third beam source for the plan of longer H-mode discharge in KSTAR. Also, the multi-beam aperture accelerator of two beam sources is being modified to enhance the beam transport efficiency by beam steering from the outermost beam apertures. Goal of the KSTAR second phase operation requires new off-axis beamline to develop the advanced operation modes (hybrid and steady state operation mods) since the broad current profiles peaked off-axis are beneficial for steady-state high performance using off-axis NBI in DIII-D experiments [1]. Therefore, the second NBI for KSTAR is under design to have the off-axis beamline of upper and lower beams steered in vertical plane. This paper presents the operation performance and upgrade of the first NBI system and the design results of new off-axis NBI with integrated tokamak modeling code for the hybrid and steady state operation. [1] D.N. Hill, Nucl. Fusion 53 (2013) 104001.

Id 443

Abstract Final Nr. P4.024

RF conditioning test and simulations on a Vacuum Feedthrough for high-power ICRF operation at KSTAR

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A high-power coaxial vacuum feedthrough (VFT) for the KSTAR ICRF antenna was designed, fabricated and tested. It utilizes two cylindrical vacuum breaks formed of an alumina ceramic (Al2O3). Two cylinders are coaxially disposed between tapered coaxial conductors to form a vacuum sealed connection between a pressurized coaxial transmission line and an ICRF antenna located within a vacuum container of KSTAR. A VFT is one of the most critical parts of an ICRF antenna. It must transmit RF power while keeping the antenna in high vacuum, and support antenna elements against electromagnetic forces and thermal stress. Frequently, high-power pulsed ICRF experiments at KSTAR have been limited by breakdowns at the VFT. Originally, the VFTs were silver coated for reduced rf loss. After 2013 KSTAR campaign we removed silver coating from eight VFTs and repaired the damaged surfaces. In order to study a higher standoff capability, we tested and conditioned the repaired VFTs in the vacuum chamber with a high power RF test stand. We also utilized the fast interlock circuit to reduce arc energy. The fast interlock can turn off the rf transmitter within 400 ns. The amplitude of rf breakdown voltage of the conditioned VFT is measured to be ~55 kV in the normal vacuum pressure of 1~5*10-5 mbar. The breakdown voltage is not changed severely up to 5*10-3 mbar. Breakdown seems to occur along the ceramic surface. In order to investigate this phenomenon we have performed simulations on the ICRF VFT using a particle-in-cell code, MAGIC2D. This paper will show the high power rf test results and upgraded design to increase both reliability and long pulse rf transmission capability.

Id 177

Abstract Final Nr. P4.025

RF design and tests on a broadband, high-power coaxial quadrature hybrid applicable to ITER ICRF transmission line system for load-resilient operations

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RF design and network analyzer tests of a broadband, amplitude-balanced quadrature hybrid are presented. We have designed two 3 dB hybrid splitters with 9 and 12 inches coaxial transmission lines applicable to ITER ion cyclotron resonance of frequency (ICRF) for load-resilient operations using high frequency structure simulation (HFSS). Amplitude-balanced wideband responses were obtained with the combination of impedance reductions of longitudinal and transverse branches in unequal proportion, length change of 50 ohm lines and diameter change of high impedance lines connected transverse to the T-section of the hybrid splitter, respectively. We have also fabricated and RF tested a 9 inches coaxial broadband hybrid coupler. We obtained an excellent coupling flatness of -3.2 ± 0.2 dB, phase difference of 4 degree and return loss of 16 dB in 40-55 MHz. The measured data of 9 inch coaxial hybrid splitter are highly consistent with HFSS simulations. We found that the proposed 3 dB hybrid splitter can be tunable with amplitude-balanced, broadband response by changing dielectric insulators to keep the inner and outer conductors of coaxial line apart. The proposed broadband, 3 dB hybrid circuit with significantly flat amplitude-balanced response can be utilized for load-resilient operations in a wide range of antenna load variations due to mode transitions or ELMs at fusion plasmas.

Id 177

Abstract Final Nr. P4.026

Status and progress of KSTAR LHCD system upgrade

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Lower hybrid current drive (LHCD) system has been installed at KSTAR since LHCD is known as a crucial actuator for the advanced tokamak operation regime. In the KSTAR LHCD system, 5 GHz microwave generated by the klystron (made by TETD) is guided in the 80 m long transmission line consisting of WR284 oversized waveguides for TE10 mode with reduced resistive loss. The transmitted power is divided into eight by the triple bifurcation using seven magic tees in the power dividing network and then fed into eight four-way-splitters which form 4R x 8C grill type launcher. The waves launched from adjacent columns have 90 degree phase shift and the consequent N// spectrum is expected to have a main peak at 1.9. The LHCD system was first commissioned during 2012 campaign. Klystron, high voltage power supply, RF control, and others including interlock system were operated successfully, however, it was difficult to apply long pulse of high RF power since the pulses were chopped by interlock which was triggered by arcs and large reflection due to wrong phase shift in the antenna. The phase problem was corrected by inserting new phase shifter and the ALOHA calculation predicted that the reflection would be reduced. Argon gas diffusion through the silicon rubber O-ring from pressurized waveguide to KSTAR vacuum was detected during pressurization test to evaluate the leakage of the waveguide flange assembly. Although it was expected that the diffusion of SF6 molecule would be much smaller than the inert gas, LHCD transmission line had to be evacuated for the sake of the secured KSTAR vacuum. In this paper, it will be described the improvement since the first commissioning, the status of the KSTAR LHCD system such as vacuum sealing of the transmission line, installation of the LHCD-dedicated-Hard X-ray camera loaned from CEA for fast electron Bremsstrahlung measurement.

Id 669

Abstract Final Nr. P4.027

Power Supply System for KSTAR Neutral Beam Injector

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The power supply system in KSTAR neutral beam injector consists of low voltage and high current DC power supplies for plasma generator of ion source and high voltage and high current DC power supply for accelerator grid system. The arc discharge is initiated by an arc power supply supplying the arc voltage between the chamber wall and twelve filaments which are heated by individual filament power supply. The negative output of arc power supply is common to each positive output of twelve filament power supplies. To interrupt the arc discharging for the fault condition of the arc current unbalance, DCCT current monitor is placed at the positive output cable of the filament power supply. The plasma grid(G1) power supply has the maximum capability of 120kV/70A which consists of low voltage regulator with IGBT-switched chopper array system for the voltage control in unit of 600V and the high voltage rectified transformers to supply DC voltage of 20kV, 30kV, and 50kV. The output voltage of the G1 power supply is also connected to the input of the voltage divider system which supplies the gradient voltage to the gradient grid(G2) in the range of 80-90% of G1 voltage by changing tap of winding resistors in unit of 1%. The charged G1 voltage is turned on and off by the high voltage switching(HVS) system consisting of MOSFET fast semiconductor switches which can immediately be opened less than 1 microsecond when the ion source grid breakdown occurs. The decelerating grid(G3) power supply is inverter system using capacitor-charge power supply to supply maximum -5kV/5A. The important component in power supply system is the surger absorber near the ion source to limit the arc energy deposited to accelerator grid. This paper presents configuration and features of power supply system, main controller, and interlock system of KSTAR NBI.

Id 496

Abstract Final Nr. P4.028

Field-Aligned-Impedance-Transforming ICRF antenna for LHD

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A pair of ICRF antennas was fabricated and installed in the LHD. The features of the antennas are field-aligned structure and impedance transformer in the transmission line. Therefore, we named the new antennas field-aligned-impedance-transforming (FAIT) antennas. These antennas are designed based on HAS antennas in LHD. The main differences of antenna heads are length and thickness of center strap. The length is about half of that of HAS antenna in order to reduce voltage on the strap. The thickness is twice compared to that of HAS antenna to reduce the convergence of electric field on the edge of strap. The loading resistance of FAIT antenna is deduced to be too low since the loading resistance of HAS antenna is low and the FAIT antenna head is shorter than that of HAS antenna. In order to increase the loading resistance, impedance transformer was designed. Heads of HAS antennas are tilted by 12?to reduce the electric field parallel to magnetic field line, which causes the formation of RF sheath. However some Faraday shields are crossing the magnetic field. On the other hand, FAIT antennas have perfect field-aligned structure. To investigate the performance of impedance transformer, we compared loading resistance of HAS antenna and FAIT antenna by injecting RF power into the helium plasma with hydrogen minority ions. The loading resistance of FAIT antenna was 2.5 to 3 times higher than that of HAS antenna in spiteof shorter strap length. Another purpose of impedance transformer is to protect ceramicfeed-through. We compared temperature on ceramic feed-throughs between HAS and FAITduring long pulse operation. Temperature of FAIT was about one third of HAS temperature. Feed line was simulated with HFSS code using experimental antenna impedance.The voltage on the feed-through is low and any convergence of electric field was not seen.

Id 378

Abstract Final Nr. P4.029

Numerical investigation of sub-cooled flow boiling effects in the collector of a 1MW ITER gyrotron operated with vertical sweeping

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In ITER, the Electron Cyclotron Heating and Current Drive (ECH&CD) system will be made of 24 gyrotrons - operating at 170 GHz with an output power of ~ 1 MW and having an efficiency of approximately 50%. Under such conditions the remaining ~1 MW of power from the electron beam is to be exhausted by the gyrotron collector, which is a vertical hollow copper cylinder, cooled by sub-cooled water in forced circulation onthe external surface of the collector. One solution adopted to enhance the heat transfer on the wetted surface of the collector is to equip it with annular fins. In this study we consider the case of a collector wall equipped with rectangular fin cavities with aspect ratio 3 and subject to vertical magnetic field sweeping of the spent electron beam. The commercialsoftware STAR-CCM+® is used to perform the thermal-hydraulic numerical analysis of the collector. The heat flux deposed by the swept ebeam is computed by detailed electron trajectory simulation with the code Esray, and is applied to the inner side of the collector. Notwithstanding the smoothing of the heat load provided by the thermal diffusion in the copper, it is shown that, at the locations of the load peaks (several MW/m 2), boiling is induced in some of the cavities. The VOF-Rohsenow boiling model predicts that the vapor remains, however, trapped in the bottom of the cavities, where a flow structure of three vortices is computed. Local film boiling takes place at few limited locations. In the paper the computed heat transfer coefficients in the cavities are presentedas well as the computed temperatures in the copper structure. In case of vertical sweeping only, which is rather pessimistic compared to the combined (vertical + transverse) sweeping adopted as reference operating scenario in ITER, the peak temperature in the copper, is close to the design limit.

Id 423

Abstract Final Nr. P4.030

Development of a high-current ion source with slit beam extraction for neutral beam injector of VEST (Versatile Experiment Spherical Torus)

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A high-current pulsed ion source has been developed for a neutral beam injector (NBI) of VEST (Versatile Experiment Spherical Torus) which is for high-beta fusion plasma experiments in the Seoul National University. The ion source is based on electron-gun assisted plasma, and ion beam extraction system employed triode configuration with slit extraction apertures will be used to study characteristics of merging and bending ion beams in the transport region with external magnetic field. To obtain positive hydrogen ion beam of 20A current at 20-30 kV energy for 10 ms, high-voltage pulse power system for the electron-gun and beam extraction system is prepared. Uniformity of the plasma source is optimized by changing number of electron-guns, and ion beam optics is studied for the extraction system by using an ion beam simulation code in order to reduce beam aberration in the extraction region. In this paper, detailed design of the ion source and the first ion beam extraction results will be presented. Based on this result, requirements of the neutralizer is determined in considering of the neutralization efficiency and beam accessibility to the main plasma of VEST.

Id 972

Abstract Final Nr. P4.031

Progress of the 2MW/3.7GHz/2s LHW system on HL-2A

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The Lower Hybrid Wave (LHW) system is one of the major additional heating methods for magnetic confined devices. LHW waves experimentally exhibit high current drive efficiency and hence work as suitable means for controlling the current profile. For the HL-2A tokamak, a new LHW system is being constructed. The system parameters are as follows: the total power is 2MW, the frequency is 3.7GHz and the pulse duration is 2s. The LHW system layout and also its schedule is presented, which includes the recent development and status of the RF source, the transmission line and the PAM launcher. The RF source is composed of four klystrons, each one has two outputs. There are 8 transmission lines in total. The RF launcher is fed by 16 ports. It has 16 active and 17 passive grids in each row. The RF source - klystrons have been delivered in Jun. 14th, 2013 and the magnet arrived in Dec. 2013. Now the test bed is being installed to test the klystron and magnet. The equipments of the klystron have been ready, such as RF driver, oil tank, support, auxiliary power supplies, data acquisition system, control system and protection system. The key components and standard waveguide for the transmission line have been finished. The key components include circulator, water load, DC break, directional coupler, 3-dB power divider and special waveguide. One transmission line has been installed preliminarily to test the transmission character. The length of the test transmission line is more than 12m, the transmission efficiency is more than 83%, and the phase difference between 3-dB power divider's two outputs is 86.4°. Theoretical design of the PAM launcher has been finished, in which the N// is 2.75 and RC is less than 1%. It has 4×16 active waveguides and 4×17 passive waveguides, capable of delivering a total power of 2MW. Now the construction of launcher, flanges, connecting section, RF windows and bends are in process and will be finished in Apr. 2014. Test bed is being constructed to test klystron and transmission line.The schedule of this project: system installation from Jan. to Mar in 2014, system test from Apr. to May in 2014 and system injection in summer.

Id 321

Abstract Final Nr. P4.032

Development of a hydrogen negative ion source by sheet plasma

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Stationary production of negative ions are important to play an essential role in Neutral beam injection (NBI). Cesium seeded Surface-production of negative ion sources are used for NBI. However, Cesium seeded surface- production of negative ion sources are not desirable from the point of view of operating steady state ion sources. We carried out the development of negative ion sources by volume-production in hydrogen sheet plasma [1]. The sheet plasma is suitable for the production of negative ions, because the electron temperature in the central region of the plasma as high as 10 – 15eV, whereas in the periphery of the plasma, a low temperature of a few eV of obtained. The hydrogen negative a ions density were detected using an “omegatron” mass analyzer, while the electron density and temperature were measured using a Langmuir probe. Negative ions current extracted from the grid are measured by Faraday-cup. [1] A.Tonegawa, K. Kumita, M. Ono, T. Shibuya, K. Kawamura, Characteristics of Hydrogen Negative Ions in Sheet Plasma, JJAP, 45 (2006)8212.

Id 954

Abstract Final Nr. P4.033

Design of a Remote Steering Antenna for ECRH Heating of the Stellarator Wendelstein 7-X

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For the Stellarator Wendelstein 7-X, which is currently under construction in Greifswald/Germany, a 10 MW 140 GHz ECRH heating system is installed. In addition to the front steering launchers, there are ports in the vacuum vessel, which allow ECRH injection from the quasi high field side. These ports, however, are too narrow for a front steering system with movable mirrors near the plasma. This leaves a remote steering design as the only option for these 2 ports. The principle of remote steering antennas is based on the imaging characteristics of corrugated rectangular waveguides, which is well understood and can accurately be simulated. Several details, however, require deeper investigation. Due to space limitations, the antenna cannot be straight but needs a mitre bend. For safety reasons, a vacuum valve, which requires a 22 mm wide gap in the waveguide, is necessary. The positions of the mitre- bend and the gap need to be chosen carefully to minimize the degradation of the imaging properties and the stray radiation in the valve. Furthermore, investigations are done to improve steering range, which is limited to approx. ±12° for a rectangular waveguide. By slighty deforming the cross section, the dispersion relation of the modes can be changed in order to increase the steering range. The simulation of the beam propagation in such an optimized waveguide requires the precise calculation of the fields and resonance frequencies of the eigenmodes. This is done with a dedicated FDTD code and a commercial tool. The antennas are manufactured from copper by electroforming. This allows to integrate all components, including the corrugated inner walls and the cooling channels, in one vacuum-tight piece. This paper will review the design process of the remote steering antennas for W7-X as well as technological issues and experimental results from test pieces.

Id 774

Abstract Final Nr. P4.034

Advanced Magnetic Divertor Control on DIII-D

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Control of two proximate X-points in the plasma divertor region is demonstrated simultaneously at DIII-D to enable advanced magnetic divertor configurations that manage the heat flux at the divertor plate. Advanced magnetic divertor configurations such as the Snowflake Divertor (SFD) have many advantages in terms of divertor target heat flux management for future fusion reactors compared to the standard divertor. The SFD creates a second-order null-point by bringing together two first-order null-points of the standard divertor. SFD geometry greatly reduces peak heat flux through its high poloidal flux expansion, a large plasma-wetted area and extra strike points. However, the SFD topology is unstable and requires complex magnetic control. We implemented the world's first real-time snowflake detection and control system on DIII-D in order to stabilize this configuration. The algorithm calculates the position of the two null-points in real-time by locally expanding the Grad-Shafranov equation. The effects of the variation in the Polodial Field (PF) coils on the change in the location of the two null-points are analytically derived. This formulation enables simultaneous control of multiple PF coil currents to achieve the requested distance and angle of the SFD. This control enabled SFD minus, SFD plus and exact SFD configurations with varying separations in scenarios such as the Advanced Tokamak discharges in DIII-D. This led to a 2.5x increase in the flux expansion and a 2.5x reduction in peak heat flux for many energy confinement times (2-3 s) without any adverse effect on core plasma such as confinement. The actuator and diagnostic requirements for future fusion reactors to enable robust control with advanced magnetic divertor configurations are discussed. *This work was supported by the US Department of Energy under DE-AC02-09CH11466, DE-AC52-07NA27344, DE-FC02-04ER-54698, and DE-AC05-00OR22725.

Id 653

Abstract Final Nr. P4.035

Divertor plasma shape reconstruction from two kinds of magnetic sensors and eddy current effect on QUEST

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In the spherical tokamak QUEST (Bt = 0.25 T, R = 0.68 m, a = 0.40 m), as one of the methods to obtain a steady-state divertor plasma, a high-density divertor plasma is made by OH (ohmic heating) and the plasma current is planned to be sustained by EBW current drive. OH divertor plasma of lower triangularity (Candy-shape) was produced with PF35-12 inner and divertor coils connected in series. And the one of higher triangularity (D-shape) was produced with PF35-1 inner divertor coil. The divertor plasma was designed by TASK/EQU code and the plasma boundary shape was reconstructed by CCS (Cauchy Condition Surface) method according to data from two kinds of magnetic sensors (flux loops, magnetic probes and partial Rogowski). Though the latter magnetic probes and partial Rogowski detect local magnetic field, the reconstruction has become possible with the standard deviation and eddy current effect adjustment. The reconstructed result showed a double-null divertor configuration and was consistent with that by E-FIT code. In the present OH plasma with a lot of high-energy electrons, there may be anisotropic plasma pressure, which makes difficult a usual equilibrium analysis, but the CCS method can reconstruct the plasma shape precisely regardless of the anisotropy [1,2]. Since a lot of magnetic probes have been installed in addition to flux loops inside the vacuum chamber, CCS can be set on the measuring (magnetic sensor) surface. Vacuum vessel and the outer space are also outside of vacuum region. Boundary integral equation is applied also on the magnetic sensor surface. Eddy current and PFC do not have to be considered in this case. [1] K. Kurihara, Fusion Eng. Design, 51-52, 2000, pp.1049-1057. [2] K. Nakamura, Y. Jiang, X.L. Liu, O. Mitarai, et al., Fusion Eng. Design, 86, 2011, pp.1080-1084.

Id 776

Abstract Final Nr. P4.036

Study on K-DEMO steady-state operation scenario and system code devleopment

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Recently, strategic plans on the high magnetic field fusion device K-DEMO (Korean Demonstration Fusion Reactor) have been published. Korean system analysis code package has been developed as the preliminary version of the tokamak simulator, focusing on the integrated current drive analysis. K-DEMO operation scenario is studied for the steady-state issue and to reduce recirculating power. An integrated numerical system is established on a platform structure. This structure provides the framwork for the whole numerial system. 0d system analysis code, 1d transport code, and 2d equilbrium & current drive code are implemented as a sub-module on the platform framework to make an integrated numerial package. 0d system code analyze (a) conceptual design parameters and (b) plasma operation regime. Input and output variables of every code are standardized for efficienct data interaction among the calculation modules. Code modules can be turned on/off depending on the calculation scheme, accuracy, and purpose. Data management nodes are used to store and redistriubute the physics output results in the platform. The Korean system analysis code package is used to suggest a current drive scenario for K-DEMO. Current drive location, efficiency, and technological readiness are considered. Exploration of self-consistent heating and current drive scenario showed the possilbility of the steady-state operation with low external power. By comparing the latest current drive technology and the suggested operation scenario, the direction of progress in the current drive technology is proposed.

Id 979

Abstract Final Nr. P4.037

Design of Discharge Waveform on HL-2M

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HL-2M is the upgrade version of HL-2A. The plasma current will reach to 3MA with the toroidal magnetic field (Bt) about 2.2 Tesla, while the major radius, minor radius, elongation and triangularity will be up to 1.78 m, 0.65 m, 1.8 and 0.7, respectively. The Equilibrium FITting code (EFIT) has been used to calculate equilibrium configuration on HL-2M. In this paper, the first part briefly introduces the main parameters of CS and PF, which are up-down symmetry, and located between Toroidal Field (TF) coils and Vacuum Vessel. These PF coils are called shaping coils for divertor configurations. In the second part, the three basic equilibrium configurations (LSN, DN and limiter) have been designed by EFIT, and the influences of β_p and l_i on equilibrium have been studied. In addition, these equilibrium configuration results from EFIT are benchmarked with those from SWEQU by other research group in Southwestern Institute of Physics. In the third part, three basic discharge waveforms with the plasma current equal to 1.2 MA and 3 MA have been designed. In our design model, PF coils provide shape control capacity as well as volt-second consumption. The CS coils together with PF coils can provide about 17 volt-seconds, which maintain 5 seconds flattop for 3MA plasma current. The last part will give a summary.

Id 253

Abstract Final Nr. P4.038

Scenario Development of First Plasma for HL-2M

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In order to address ITER relevant physics issues, HL-2A tokamak was being update to HL-2M. A practical model is being developed for the first plasma initiation on HL-2M, which is based on the valuable start-up information provided by the present tokamaks, JET, JT-60U, and DIIID. The model can optimize the field null configuration, compute the loop voltage for plasma breakdown, simulate the evolution of the eddy current on the vacuum vessel and the waveforms of the PF coils, and design the algorithms of plasma control for different discharge phases.

Id 494

Abstract Final Nr. P4.039

The Design and Development of Big Control System for HL-2A/HL-2M

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Big control system for HL-2A/HL-2M is not a new control system which will be built from scratch but to reorganize all of the current control subsystems and combine these subsystems into one from the perspective of large-scale system using a series of method. The ultimate goal of the big control system is efficiency of the operation and control for HL-2A and for HL-2M in the future. In our design, informatization, automation, intellectualization are the basic characteristic for big control system. For the informatization, we setup five standards for all subsystem involved the operation and control, which are a reflective memory network, PLC ring network, unified timing network, EPICS network and HDF5 data format, so that all the subsystem can connect each other using standard interfaces. And the big control system also provides several interfaces for human such as SMS, voice, email, phone and instant message to help the remote participation and cooperation, all-round display, off-site duty etc. For the automation and intellectualization, there are three aspects have been considered in the big control system. The first is the automation driving sources such as network interrupts in reflective memory network for real-time calculation and the EPICS message for the normal calculations performed in the gap between shots. The second is the modified codes and the algorithm such as EFIT,TSC which are suitable for the automatic operation. The last is emerging technology such as big data, expert systems and artificial intelligence technology based on the vast amounts of historical data and operating experience to predict the consenquence and do a comprehensive analysis for engineering failure and physical phenomena This work is supported by the Chinese National Fusion Project for ITER under Grant No. 2012 GB105001.

Id 492

Abstract Final Nr. P4.040

The analyzation of a new disruption mitigation method based on ETC in J-TEXT

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To analyze the dissipation process of the poloidal magnetic energy of the plasma during the current quench phase (CQ), a transformer model is built for the J-TEXT Tokamak. Which clearly shows that the magnetic energy is dissipated totally by radiation due to ohmic heating of the cold resistive plasma. This result coincides with simulation results of the Tokamak simulation code (TSC), and the experiment results by calculating the energy absorbed the plasma through the ohmic heating. On the other hand, to reduce energy consumption left in the vacuum vessel led by the radiated poloidal magnetic energy, a new disruption mitigation method based on the energy transferring coil (ETC) is explored. Only a switch is seriesed with the ETC. The ETC coupled with the plasma, is put into use only during the CQ by closing the switch and forming a loop. The ETC can transfer the plasma magnetic energy out of the Tokamak device through electromagnetic coupling and the energy consumption left in the vacuum vessel would be reduced. Which is a great advantage of the ETC over the existing mitigation methods, such as gas injection. The primary simulation results show that about fifty percent of the magnetic energy is transferred when the number of the turns of the ETC is ten and the resistance of the ETC is ten milliohm.The induced current in the ETC can also compensate the rapidly reduction of the poloidal magnetic field caused by the current quench and limit the toroidal electric field, which may also have an effects of runaway electron suppression. In conclusion, the ETC can be studied as a new method of disruption mitigation and can cooperate with the existing mitigation methods to protect the fusion device.

Id 584

Abstract Final Nr. P4.041

Improvements in Disruption Prediction at ASDEX Upgrade

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In large-scale Tokamaks disruptions have the potential to create serious damage to the facility. Hence disruptions must be avoided, but, when a disruption is unavoidable minimizing its severity is mandatory. A reliable detection of a disruptive event is required to trigger proper mitigation actions. To this purpose machine learning methods have been widely studied to design disruption prediction systems on several experimental devices. In particular, for ASDEX Upgrade, some of the authors presented predictive systems applying data based techniques, such as Multi-layer Perceptron neural network [1], Discriminant Analysis [2], and Self-Organizing Maps [3]. The training phase of the proposed approaches is based on the availability of disrupted and non-disrupted discharges. To accomplish an exhaustive model every disruptive and safe configurations included in the machine operational space should be represented in the training set. Safe configurations were selected from safe discharges, while disruptive configurations were assumed appearing into the last 45ms of each disruption [1,3]. Even if the achieved results in terms of correct predictions were good, it has to be highlighted that the choice of such a fixed temporal window might have limited the prediction performance. In fact, it generates ambiguous information in cases of disruptions with disruptive phase shorter than 45ms. Conversely, missing information is caused in case of disruptions with a disruptive phase longer than the prefixed one. The assessment of a specific disruptive phase for each disruptive discharge represents one of the most relevant issues in understanding the disruptive events. Several similarity measures, such as Mahalanobis distance, and statistical methods, such as Logistic Regression, have been applied to evaluate the membership of each sample to the safe or the disruptive configurations. Preliminary results show that enhancements on the achieved performance on disruption prediction are possible by defining a specific disruptive phase for each disruption. [1] B. Cannas et al. 2010 Nucl. Fusion 50 075004 [2] Y. Zhang et al 2011 Nucl. Fusion 51 063039 [3] R. Aledda et al. 2012 IEEE Trans. On Plasma Science 40, no.3, 570 – 576.

Id 534

Abstract Final Nr. P4.042

A novel approach for solving three dimensional eddy current problems in fusion devices

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In devices aiming at magnetic confinement of fusion relevant plasmas, time varying magnetic fields induce eddy currents in the conductive structures (e.g. vessel, blanket modules , etc.). In this paper we present an original approach to efficiently compute 3D eddy currents by exploiting any geometrical symmetry of the machine. For example, ITER's conducting structures can be modelled by gluing 9 pieces of the same structure (symmetry cell). Therefore, one may solve 9 independent problems derived from spatial Fourier analysis of the field sources on 1/9 of the whole machine, providing a drastic reduction of the computational time. Moreover, since the symmetry cell still exhibits a mirror symmetry, a more efficient—but more complicated—technique can be exploited. Then, the symmetry cell becomes 1/18 of the whole structure. On the other hand, the complication of using dihedral symmetry is that this group is non-Abelian, which implies that the 18 complex valued sub-problems—one for each symmetry cell—are not independent but in most cases coupled in pairs by boundary conditions on the symmetry boundary. The aim of this paper is to effectively implement in the electromagnetic code CAFE [1] the recipe introduced by Bossavit in [2]. The novelty is an original technique to solve the coupled sub-problems by using a domain decomposition approach, which increases noticeably the efficiency of the simulation. The proposed technique can be applied also in the presence of field sources that do not necessarily share this symmetry, as for example for Asymmetric Vertical Displacement Events (AVDEs). [1] P. Bettini, L. Marrelli, R. Specogna, Calculation of 3D magnetic fields produced by MHD active control systems in fusion devices, IEEE Trans. Magn., 2014, DOI: 10.1109/TMAG.2013.2279141 [2] A. Bossavit, The exploitation of geometrical symmetry in 3-D Eddy currents computation, IEEE Trans. Magn., Vol. 21, No. 6, pp. 2307–2309, 1985.

Id 429

Abstract Final Nr. P4.043

Table-top pelletinjector (TATOP) for impurity pellet injection

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A table-top pellet injector (TATOP) has been designed to fulfill the following scientific aims: to study the ELM triggering potential of impurity pellets, and to make pellet injection experiments comparable over several fusion machines. The TATOP is based on a centrifugal accelerator therefore the complete system is run in vacuum, ensuring the compatibility with fusion devices. The injector is able to launch any solid material in form of balls with a diameter in the 0.5 – 1.5 mm range. The device hosts three individual pellet tanks that can contain e.g. pellets of different material, and the user can select from those without opening the vacuum chamber. A key element of the accelerator is a two-stage stop cylinder that reduces the spatial scatter of pellets exiting the acceleration arm below 6°, enabling the efficient collection of all fired pellets. The injector has a maximum launch speed of 450 m/s. The launching of pellets can be done individually by providing TTL triggers for the injector, giving a high level of freedom for the experimenter when designing pellet trains. However, the (temporary) firing rate cannot be larger than 25 Hz. The controlling of the device is implemented using two microcontrollers, one of them hosting a web server, enabling the user to control the injector remotely, and the other one fully devoted to device control and safety. To detect malfunctions, the TATOP is equipped with a 3D resonance detector, as well as the vacuum pressure, arm rotation speed and power consumption are monitored, which all can generate an emergency shutdown. In this paper, besides the injector design, laboratory test results and a real-life application example at ASDEX Upgrade tokamak is presented. The AUG experimental results are expected to be available before the conference.

Id 411

Abstract Final Nr. P4.044

Cyclic data processing approach for long-pulse massive data analysis in real time

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Frequency modulation (FM) Reflectometer has been developed to measure the plasma density profile of the KSTAR. Three reflectometers (Q, V, and W band) are configured with PXI based digitizers and arbitrary function generators. The mixer output of Reflectometer is directly digitized with a sampling rate of 100Msamples/s. Digitized data are archived on the on-board memory temporarily during discharge experiment, then, are transmitted to the central archiving server. Due to the memory limit of 32Mbytes, the data are recorded only for 0.16s in a shot. Hence, the many - triggering method is used to measure signal only in the interesting period of plasma discharge. As the pulse length increased, a novel method is required to get the entire data during the whole pulse period. Our preliminary design focuses on streaming data transmission from local device to a central server. Continuous data transmission scheme is implemented as a cyclic data acquisition system (C-DAS) which is synchronized with low speed input clock. The FPGA based C-DAS is able to operate with various system configurations, e.g. we can configure it to have 100Mhz sampling speed with very short time, 1ms, and data acquisition logic is iterated with the 10Hz rate during whole pulse operation. Digitized data are transmitted to the host side memory directly by means of the DMA engine in FPGA. We expect this cyclic data acquisition function is available to different diagnostic system that need ultra high-frequency data acquisition. Therefore, C-DAS is configured with the microTCA.4 hardware platform. The proposed approach will increase the system reliability and flexibility in KSTAR.

Id 692

Abstract Final Nr. P4.045

New development of epics based data acquisition system for H-Alpha diagnostic.

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The H-Alpha diagnostic system has been developed to measure the line integrated intensity in the direction of toroidal and poloidal. The DAQ system for H-Alpha diagnostics at the beginning of the first plasma in 2008 was developed with VME-form factor digitizer in the Linux OS platform. The VME digitizer module of H-Alpha DAQ system was modified to measure the low current signal from the PMT. The input maximum current values of modified digitizer module are 400nA and low current data is expressed as the value of the voltage between -10V and +10V. At first time, there was no problem to measure KSTAR H-Alpha signal, but it couldn't measure the H-Alpha data signal as the KSTAR plasma density increased. It exceeds digitizer input range, which means the H-Alpha signal is over 400nA, so we should change the resistor on the digitizer board to measure the 400nA over current. But it is not easy to change resister on the digitizer board. And it showed unstability in the long time operation with high sampling data acquisition. In order to overcome these weak points, a new H-Alpha data acquisition system has been developed with a CPCI based digitizer and a signal conditioning box for converting the current to voltage in the Linux OS platform. The new DAQ system was developed based on EPICS framework like other KSTAR diagnostics with SFW(Standard software framework). The main advantages of the H-Alpha DAQ system are the capabilities of calculating line integrated intensity during the plasma shot and displaying it in run-time. During the plasma shot, the new DAQ system performs segment archiving data in MDSplus DB of KSTAR Central Storage.

Id 916

Abstract Final Nr. P4.046

Breadboarding and thermal testing of the first mirror unit for H-alpha and visible spectroscopy in ITER

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H-alpha and visible spectroscopy (HA&VS) diagnostic in ITER is aimed at the measurements of intensity of hydrogen isotopes and of the impurity lines in the SOL emission. It is based on the endoscope scheme with first mirror unit (FMU) containing two mirrors (flat and aspheric) located behind the small entrance pupil. The FMU main function is to provide an image of the proper part of ITER first wall with sufficient quality and stability under thermal, radiation and electromagnetic loads. It is planned to combine it with a cleaning system for periodic recovery of mirror surface by a discharge, if the essential contamination occurs. Earlier experiments have shown that single-crystal molybdenum (SC-Mo) is the best candidate material for the FMU mirrors, since SC-Mo reflection properties do not degrade substantially under surface sputtering by high-energy particles. The maximum FM size for HA&VS is up to 200 mm and higher. Unfortunately, the sufficiently large SC-Mo blanks are commercially unavailable. A sandwich-like design is proposed, in which few small-size SC-Mo plates are bound by a diffusive welding to a large-size polycrystalline Mo substrate. Options on bonding technique and sandwich SC-Mo mirror designs are presented. Recent progress is discussed on the polishing of SC-Mo focusing mirrors up to 150 mm diameter. Thermal tests of the molybdenum FMU breadboard had been performed in a vacuum chamber with windows for optical measurements. The FMU was heated up to 250°C whereas the mirror temperature was up to 350°C under the pressure of 8-9 Pa. A technique for quantitative assessment of an image quality (contrast and resolution) at elevated temperatures has been developed. As a result, a sub-mm shift of an image plane along the optical axis had been observed with no significant degradation of the image quality.

Id 447

Abstract Final Nr. P4.047

Thermal deformation analysis of the first mirror unit for tangential-view EP12 channel of H-alpha and visible spectroscopy in ITER

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H-alpha and visible spectroscopy (HA&VS) diagnostic in ITER is supposed to play a primary role for a number of measurements derived from detected intensity of the Balmer lines of hydrogen isotopes and of the impurity lines in the SOL emission. First mirror unit (FMU) is a critical component of any optical diagnostics in ITER being subjected to most powerful radiation, thermal and electromagnetic loads. The front surface of first mirror (FM) will be exposed also to the direct fluxes of plasma radiation, sputtering by charge-exchange atoms and deposition of impurities sputtered from the vacuum vessel plasma-facing components, most likely by Beryllium from the ITER first wall blanket modules. The HA&VS diagnostic optical system is based on the endoscope scheme with FMU containing two mirrors (flat and aspheric) located behind the small entrance pupil limiting the FM exposure to the charge-exchange neutrals from the plasma. The FMU main function is to provide an image of the proper part of FW and transfer it to the vacuum window with sufficient quality and stability under thermal, radiation and electromagnetic loads. Also, it is planned to combine it with a cleaning system for periodic recovery of mirror surface by a discharge, if the essential contamination occur. Earlier experiments have shown that single-crystal Molybdenum (SC-Mo) is the best candidate material for the first mirrors, since SC-Mo reflection properties do not degrade substantially due to surface sputtering by high-energy atoms from the plasma or by ions from the cleaning discharge. For the FMUs of H-alpha equatorial channels, expected maximum thermal load caused by the nuclear heating is approximately 0.5 W/cm3 in Molybdenum. The ANSYS analysis for thermal deformations of the FMU design carried out for EP12 tangential-view channel is present, followed by the evaluation of its impact on the optical performance by ZEMAX code. The maximum FMU temperature of ~300°C has been derived resulting to micrometer-scale deformations which cause minor degradation of an image quality within acceptable limits. Small displacements of image plane and chief ray direction could be easily corrected by the proper adjustment of detector position.

Id 620

Abstract Final Nr. P4.048

The science and technology in the design of the ITER Diagnostic Residual Gas Analyzer

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The US-ITER Domestic Agency project to develop the ITER Diagnostic Residual Gas Analyzer (DRGA) is currently entering its Final Design stage. The base design includes two, independent, complete systems, each integrating three measurement sensors to analyze the neutral gas composition in the main chamber and in the divertor duct. Main emphasis is placed on helium (He) concentration in the ducts, as well as the relative concentration between the hydrogen isotopes (H2, D2, T2). Measurement of the concentration of radiative gases, such as neon (Ne), is also intended. As part of ITER's plasma diagnostics set, the DRGA is designed to operate during the plasma discharge, which in ITER will have a duration of up to ~1 hour, with sufficient response time to resolve gas concentrations in scale times compatible with plasma-wall interactions (~1s for the divertor system – somewhat higher for the non- hydrogenic species and 10x higher for the main chamber). To achieve these capabilities, these systems are being engineered with physics sensors compatible with the harsh environment of the ITER port-cell, using special, radiation hardened sensors were possible and with most electronics separated from the sensors and radiation shielded. With working gas pressures as high as 10 Pa anticipated for the ITER divertor ducts, the divertor DRGA system is designed to maintain pressure differences of 3-4 orders of magnitude across its chamber. The safe positioning divertor system's analysis chamber has required a 7m long extension of ITER's primary vacuum, traversing the toroidal field cryostat, to form the sampling pipe. Nevertheless, flow calculations (covering both molecular and transition flow regimes) confirm that the ~1 s response time is maintained. A newly established DRGA prototype testing lab at ORNL has been used to confirm the sensitivity to the expected levels of He/D2 (~0.005 – 0.02, mainly produced as fusion "ash") and T2/D2 (0.01 – 10, but tested with H2/D2) and Ne/D2 (~0.01, if no significant compression) in the ITER pumping ducts.

____ †Work supported, in part, by the US DOE under Contract No. DE-AC05-00OR22725 with UT-Battelle, LLC. *Presently with the International Fusion Energy Research Centre, Rokkasho-mura, Kamikita-gun, Aomori 039-3212, Japan

Id 915

Abstract Final Nr. P4.049

Forecasting time series for recognition of anomalous behaviors in waveforms

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Predicting future response of a system or future values of a signal requires appropriate models that capture the core dynamics of a physical phenomenon on a particular temporal horizon. Although simulation and prediction could be considered as similar approaches, they are actually very different from a conceptual point of view. Simulation means computing the model response using input data and initial conditions, while prediction forecasts the model response n steps ahead into the future by using the current and past values of observed input and output values. In this sense, forecasting is much less demanding on knowledge about the system dynamics than the simulation is. In particular, plasma dynamics depends on many variables and parameters, which are still out of our overall understanding, making difficult to model it. Instead of that, we are full of massive databases with enormous amount of acquired data, which come from thousands of fusion diagnostics. Advanced algorithms could take advantage of those large databases to perform forecasting of several time series data in a short temporal horizon. These predictions could be used to anticipate, in real time, abnormal plasma behaviours during a discharge. This contribution provides a comparative study of the application of algorithms to perform forecasting of temporal evolution signals. Simple techniques such as trend models and exponential moving average or much more advanced ones as dynamic neuronal networks are considered here. The study has been applied to several waveforms of the database of the experimental fusion device TJ-II. The results have shown that depending on the temporal horizon, the prediction of future values of some signals match the observed ones in most cases with an error less than 5%. The work also discusses some particular issues of the real time implementation of the selected forecasting algorithms.

Id 818

Abstract Final Nr. P4.050

General overview of the ITER low field side reflectometer diagnostic system

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One of the diagnostic systems being provided to ITER by the US is the Low-Field-Side (LFS) Reflectometer. The LFS Reflectometer is one of several reflectometer systems planned for ITER. The LFS X-mode system, reflecting off the upper cutoff, is used to probe the on and off-axis edge pedestal and scrape-off layer (SOL) electron density (ne) profiles. While the main function of this diagnostic is the measurement of the electron density (ne) profile, it will also be used for core MHD and turbulence measurements. In addition the LFS system includes a Doppler reflectometer with oblique views to the plasma flux surface, for turbulence rotation measurements. The ‘front-end’ components of the LFS Reflectometer are housed in an equatorial port plug. The LFS reflectometer components share this plug with four other diagnostics. The current LFS Reflectometer design contains 7 circular waveguides that function as both launch and receive antennas, with penetrations through the diagnostic first wall providing access to the plasma. The millimeter waves are coupled quasi-optically to the corresponding waveguides outside the vacuum through double quartz windows. The reference design features 7 broadband multimode corrugated circular waveguide transmission lines. There are multiple miter-bends in the transmission lines as they take the signals to and from the diagnostic hall from the machine. The total length of the waveguide run from launch/receive horn to source/detector is approximately 40 meters. Detailed structural, thermal, EM and neutronic analysis of the diagnostic system have been performed which qualify the design with regard to steady state operation and during ITER plasma disruption events.

Id 216

Abstract Final Nr. P4.051

Design of Thomson Scattering Diagnostics System for VEST

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A Thomson scattering system is designed for Versatile Experimental Spherical Torus (VEST) to measure spatial profiles of both the electron temperature and density. The electron temperature and density of the ohmic plasma of VEST has been estimated to be as low as 10~200 eV and ~1x10^18 /m^3 respectively due to the low heating power capacitiy of VEST, referring the measured plasma parameters at the edge of the plasma using triple Langmuir probes. The estimated electron density is relatively low so that number of scattered photons should be relatively low as well. In order to measure plasma with sufficient signal to noise ratio even with the low number of scattered photons, the collecting optics are carefully designed to have a low F-number. A scattering length and the number of band pass filter in each polychromator are determined to increase the number of collected photons. An avalanche photodiode (APD) with low dark current has been chosen as detectors for the polychromator to increase signal level and minimize a noise. The system configurations including windows and a chord of laser are designed.

Id 950

Abstract Final Nr. P4.052

Wide-band optical coupling isolation amplifier for the Joint TEXT tokamak

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Isolation amplifiers are widely used in the subsystems of the Joint TEXT (J-TEXT) tokamak. Designing an isolation amplifier with a minimum bandwidth of 1 MHz is necessary because the sampling rate of the signals of Langmuir probe diagnostic systems is 2 MHz. A wide-band optical coupling isolation amplifier is proposed in this study. Experimental results show that the isolation amplifier can transmit an analog signal from DC to 2.7 MHz with good linearity. The propagation delay of the circuit is within 142 ns. The effects of the compensation capacitor on the frequency characteristics of the isolation amplifier were determined through simulation. A mathematical model of the circuit was established, with several parasitic parameters considered. The linearity and frequency characteristics of the circuit were studied by combining theoretical analysis, simulation, and testing. The availability of the presented isolation amplifier was verified in the Langmuir probe floating potential measurement system on J-TEXT.

Id 283

Abstract Final Nr. P4.053

A self-description data framework for Tokamak control system design

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A Tokamak device consists of numerous control systems, which need to be configured and integrated. The ITER CODAC (Control, Data Access and Communication) group has released the SDD (Self-description data) to describe the configuration of ITER plant systems. Following the ITER SDD, we developed a flexible and scalable SDD framework for J-TEXT, and it may also be applicable to other sophisticated devices. The SDD describes the configuration of various plant systems in generic models, including physical and logical elements and how they are related. The SDD models are classified into components and connections. The components represent the physical and logical elements and they are organized in a tree structure. The relationship between a parent component and children components is inclusion, meaning the child is part of the parent in physical or logical structure. Components can also be related to another through connections, indicating they exchange data or something but there is no inclusion relation. The SDD is composed of three layers: the mongoDB database, which is an open-source, dynamic schema, NoSQL database; the SDD service, which comprises an ORM (Object Relation Mapping) based on mongoDB and RESTful Web services to handle transaction and business logic; the SDD applications, which allow plant system designers to create and maintain SDD information of a specific plant system, and generate configuration files automatically.

Id 684

Abstract Final Nr. P4.054

Radiation of the megaamper Z-pinches: measurements and theoretical modeling

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One of the ways to achieve thermonuclear ignition is inertial fusion. For this goal the lasers and the megaamper generators are used as drivers. By the passing of currents through loads of a small weight the high temperature Z-pinch is formed. The pinch plasma is the source of high power radiation that can be used for compression thermonuclear target. The knowledge of radiation parameters (power, spectrum) is necessary for the compression process optimization. This investigation was carried out on Angara-5-1 facility (current up to 5 MA, rise time 100 ns). The multiwires cylindrical W and Al arrays was used as starting base of the pinch. At the final stage of the pinch compression the plasma column (diameter 1÷2 mm) is formed. This plasma is the radiation source of the soft X-ray (Ehv = 50÷5000 eV) with total yield up to 50 kJ during 5÷10 ns. For investigation of radiation spectra two spectroscopy diagnostics are used: the first on the base of transmitting grid for spectral region Ehv ≤ 1000 eV and the second spectrograph with convex mica crystal for energy quanta over 1000 eV. As detectors the special film and linear CCD are used. Results of measurements are given. The theoretical model of radiating plasma of multicharged ions which takes account levels kinetics and radiation transfer of energy was used for the analysis of experimental data. At the first stage of research the analysis was carried out by comparison of experimental spectrum with calculated one using stationary model of pinch plasmas. For this purpose the bank of spectra types in a wide range of density ne and temperature Te was created. However, the typical spectra calculated in reasonable intervals of ne and Te for the given pinch geometry considerably differ from the measured ones. The assumption was made that the divergence is connected with absence in computing model the dynamics of pinch compression. The time integrated spectrum calculated by using time resolved distribution of temperature and density with rather good accuracy coincides with the experimental one that allows to draw a conclusion on compliance of modeling to real processes. Authors are grateful to the Russian Foundation for Basic Research for support of these investigations (grants 12-01-00744, 13-02-00094).

Id 258

Abstract Final Nr. P4.055

Compact, battery powered and wireless digitizers for in situ data acquisitions in the sunist spherical tokamak

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Many modern microcontroller units (MCUs) have integrated high performance analog-to-digital converters (ADCs, up to 16 bit and 1 MS/s) and quite a few static random-access memories (SRAMs, up to 256 kB). Compact, battery powered and wireless digitizers for in situ data acquisition have been developed based on such MCUs and routinely used in the SUNIST spherical tokamak. Each wireless digitizer is powered by a Li-ion battery pack (+/-8.4V, 3100 mAH). Two seconds before a discharge, the MCU and other modules in wireless digitizers are waked up by a wireless command sent from the control center of SUNIST. When the discharge begins, the timing system sends a wireless pulse sequence, which can avoid false triggering, to each digitizers. The MCUs recognize the pulse sequence after a fixed time delay (about 30 microseconds) and start analog-to-digital conversions immediately. Tens (or hundreds, depends on the sample rate) of milliseconds of analog signals are then digitized and stored in the SRAMs of MCUs. The data stored in the SRAMs are transmitted to a data server one after another through Bluetooth or Wi-Fi networks two seconds after discharge. Except the low power consumption wireless wake-up module, all other parts of the wireless digitizers return to sleep when the data transmission is completed. Because of low duty cycle, the wireless digitizers have a battery life of 2 ~ 4 weeks and a handheld size. In general, the wireless digitizers have better performance than normal isolation amplifiers and can greatly simplified the cable connections. They are very suitable for the data acquisition of danger and/or susceptible analog signals in non-steady state small tokamaks.

Id 898

Abstract Final Nr. P4.056

Global variance reduction techniques in MCNP for ITER representative geometry

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In ITER shutdown dose rate calculations, accurate evaluation of neutron flux in extended regions far from the source (Port Interspace and Port Cell) is required, since they are of relevant interest for project requirements. To achieve this accuracy with Monte Carlo transport methods, attainment of a proper distribution of neutron fluxes in these regions of interest is necessary. Standard variance reduction techniques are unable to transport particles in all regions and techniques based on population control are necessary to effortlessly obtain accurate statistics in regions otherwise unreachable. MCNP code is able to handle population control using importance maps. The weight of the particle is related with these importance maps and, during the transport process, population is adjusted accordingly to this weight. MCNP built-in feature weight windows generator (WWG) creates weight maps optimizing a desired tally in these areas. This method promotes good statistics in a localized area but dismisses regions that don't contribute to the tally. The necessity of sampling large regions with low statistical errors requires the development of global variance reduction (GVR) techniques, where all the geometry is of interest. In this article two variance reduction methods are compared in terms of computer efficiency and sampled region, using the ITER B-lite v3 model. Both techniques are based on assigning thresholds to weight windows maps. The first one is based on an iterative process where the importance is function of the neutron flux. The second method creates the weight map using WWG over an analog run where all material densities have been reduced to a fraction of its value. Running time along with the average statistical errors and Figure of Merit over a mesh tally will be established. Since previous calculations to generate the importance map are required, time necessary to complete the whole process will also be considered.

Id 1026

Abstract Final Nr. P4.057

Integration of advanced data acquisition applications using FPGA-based FlexRIO devices in ITER's CODAC Core System

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The aim of this work is to present the development and integration of advanced data acquisition (DAQ) applications using Reconfigurable Input Output (RIO) FPGA-based devices in ITER's Control, Data Access and Communication (CODAC) Core System (CCS). CCS is the software distribution built by the ITER Organization for the developers of plant system controls. Compared with traditional DAQ systems, the use of Reconfigurable Input Output (RIO) devices drives a methodology change of the design model and brings the system designer the capability to fully customize the functionality with a high performance and a reconfigurable architecture. National Instruments (NI) FlexRIO devices are part of ITER Catalog of I&C products for Fast Controllers [1] developed using PCIe/PXIe technology. The integration of this hardware with EPICS [2][3] is relying on the common interface that the Nominal Device Support (NDS) supplies. NDS is a software layer that provides a device handling standardization in CODAC, simplifying the development of EPICS device support. The design methodology proposed in this work covers: a) modeling in LabView-FPGA the behavior of the DAQ hardware b) exporting the functionalities for interacting with the hardware to ITER's CODAC Core System (CCS) [4] c) the creation of the low level communication interface to the device using the NDS abstraction layer and the connection with the corresponding EPICS records and d) the built of the main software element of EPICS, the Input/Output Controller (IOC). Two DAQ examples using FlexRIO PXIe796x devices with analog input (5761) and digital I/O (6581) adapter modules are presented as basic use cases of integration with ITER's CCS.

Id 126

Abstract Final Nr. P4.058

RIO EPICS device support application case study on an Ion Source Control System (ISHP)

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Experimental Physics and Industrial Control System (EPICS) is a software tool that during last years has been getting relevance to be used as main framework to deploy distributed control system in big scientific environments. At the moment, ESS Bilbao uses this middleware to perform the Control of the Ion Source [1] and the Low Energy Beam Transport system (LEBT). The ion source control is included into the Ion Source Hydrogen Positive (ISHP) project which goal is, on one hand, to build a platform for test accelerators related technologies and on the other, to have an structure able to be the first step for an accelerator. The implementation of the control system was based on PXI Real Time controllers, using LabVIEW-RT and LabVIEW-EPICS tools [2]. The additional usage of an FPGA increases the capabilities of the control system. However, the control software environment did not provide a full EPICS IOC, requiring additional efforts for implementing this feature. Furthermore, the maintainability of the system might be jeopardized when many controllers are present, and the compatibility of the system with new standards relay in external vendors. Therefore, the migration of the current system to an open source software alternative, will have important improvements regarding robustness, compatibility and maintainability of the entire system. In such way, Linux operating system has been the choice to support EPICS environment in the PXI controller. This paper presents an application case study of the use of a generic RIO EPICS Device Support library [3], to give support to ISHP using a derivative Linux version of Red Hat. FPGA code is almost the same as it was before, but the use of a model design methodology provides an easy and fast deployment of new software configuration. Hence, implementing and integrating ISHP data acquisition and control using into EPICS maintaining RIO technology.

Id 624

Abstract Final Nr. P4.059

A high throughput data acquisition and processing model for applications based on gpus

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There is an increasing interest in the use of GPU technologies for real time analysis in fusion devices. The availability of high bandwidth interfaces has made them a very cost effective alternative not only for high volume data analysis or simulation, and commercial products are available for some interest areas. However from the point of view of their application in real time scenarios, there are still some issues under analysis, such as the possibility to improve the data throughput inside a discrete system consisting of data acquisition devices (DAQ) and GPUs. This paper addresses the possibility of using peer to peer data communication between DAQ devices and GPUs sharing the same PCIexpress bus to implement continuous real time acquisition and processing systems where data transfers require minimum CPU intervention. The proposed model will be tested through a generic data acquisition device. Data acquired by this system will be sent to an Nvidia Tesla GPU for real time processing. By using direct memory access (DMA) transfers, provided in the NVIDIA CUDA SDK for Linux systems, CPU usage is minimized through standard means as an alternative to third party solutions. This technology eliminates unnecessary system memory copies and lowers CPU overhead, avoiding bottleneck when the system uses the main system memory. This permits to increase the sampling rate of the acquired signals, and the possibility to run algorithms more complex in the GPU. In this scenario the CPU will only deal with supervision and control tasks, as it would not interfere with data transfers or processing. We will present the performance evaluation of this model for a data acquisition and processing application used in the context of fusion devices.

Id 570

Abstract Final Nr. P4.060

Current status of EU activities for the Remote Data Access System of the ITER Remote Experimentation Centre

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The Broader Approach (BA) is an agreement between the European Union and Japan that complements the ITER Project. The ITER Remote Experimentation Centre (REC) in Rokkasho is one of the projects currently implemented within the BA agreement. The ultimate objective of the REC is to allow researchers to take part in the experimentation on ITER from a remote location. Before ITER will be operated, the REC will test ITER-relevant technologies for remote participation on JT-60SA, which is under construction in Naka, and possibly on existing EU tokamaks. In the current plan, the overall scope of the REC activities includes the provision of software tools for Remote Data Access (RDA). However, an RDA system for ITER will require a number of features that cannot be found in any of the existing machines. In particular, the following aspects are subjects of R&D in the field: • long pulse support; remote users will require the access to ITER data while the experiment is still ongoing; • data transfer optimization over high latency networks; this will be required for the REC, given the properties of network link between EU and Japan; • data transfer optimization for data visualization tools; when RDA technologies are used by data visualization tools, specific optimization can be performed. Currently two different technologies are under evaluation for the RDA system of the REC: MDSplus and IDAM. This paper first gives a brief overview of the current status of the REC project. Afterwards, the two considered tools will be introduced. Some preliminary results about data transfer performance with different communication protocols, such as TCP/IP and UDT, will be also presented. The two considered RDA tools have been also used to make a preliminary assessment for the transfer of large data sets. A proposal for the RDA system of the REC will be then discussed.

Id 76

Abstract Final Nr. P4.061

Langmuir probes design for the actively cooled divertor baffle in WEST

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The WEST project (W-Environment in Steady-state Tokamak) aims to transform the Tore Supra limiter configuration to an X-point divertor, providing a test bed for ITER-like plasma-facing components (actively cooling W monoblocs) under high heat flux, steady state plasma irradiation. The lower divertor includes an actively cooled, W-coated CuCrZr baffle to provide neutral compression and improve particle exhaust. Baffle is made of several modules which are about 400mm long and 20mm thick. Flush-mounted Langmuir probes will be installed near the leading edge of the baffle to provide plasma flux and electron temperature measurements for physics studies and real-time machine protection functions during steady-state discharges. The baffle will be irradiated by plasma, energetic ripple-trapped ions, photons and energetic neutrals from charge exchange reactions. While not as severe as the heat loads at the divertor strike points, the baffle will be nonetheless subjected to power fluxes up to 3 MW/m2 in steady state, representing a new challenge for Langmuir probes. In this work we propose a simple, robust design that satisfies the physics requirements (probe collecting surface area and alignment tolerances) and guarantees long term survivability of the probes, under physics-based power loads. To provide good heat exhaust capability, while maintaining electrical insulation from the baffle, the CuCrZr probes are meant to be coated with a thin diamond layer and pressed into the baffle with a locked screw mechanism. Finite element thermo-mechanical analysis of the proposed design shows that high pressure thermal contact can be maintained under the worst case thermal loads. Even including the extra power load due to electrical biasing of the probes, their temperature should not exceed 390°C, which is acceptable compared to the maximum expected baffle temperature of 350°C. A final thermo-mechanical analysis has confirmed the robustness of the Langmuir design also during a Vertical-Displacement-Event (VDE).

Id 572

Abstract Final Nr. P4.062

The Absolute Wavelength Calibration of imaging X-ray Crystal Spectrometer

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As the one of the key diagnostic methods of plasma physics, imaging X-ray Crystal Spectrometer (XCS) has been widely applied on several major tokamaks, providing profiles of ion and electron temperature and the rotation velocity. However, there is a long standing difficulty for wavelength calibration due to the fact that the spectra recorded by XCS are in the soft x-ray range and appropriate standard light source was not readily available. This abstract introduces a new wavelength calibration method for the absolute rotation velocity measurement, and the detailed arrangement and preliminary wavelength calibration results of imaging X-ray crystal spectrometer on EAST are presented.

Id 596

Abstract Final Nr. P4.063

Calibration of the toroidal charge exchange recombination spectroscopy on EAST

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Charge eXchange Recombination Spectroscopy (CXRS) is currently being developed on EAST along with the commissioning of the first heating neutral beam. Besides the plasma temperature and velocity profiles, CXRS can also provide impurity density when used with beam emission spectroscopy (BES). As one major component of the EAST CXRS systems, the toroidal spectroscopy measures plasma temperature and toroidal velocity profiles. Based on the desired spatial arrangement and port accessibility, 25 channels are deployed that will cover plasma region from core to low field side with spatial resolution <4cm and expected time resolution <10ms. For the system to be useful, accurate calibration of spatial location, wavelength and absolute intensity of the toroidal CXRS is essential.My abstract forcus on different kinds of calibrations which will surely be used in 2014 EAST experiment.

Id 452

Abstract Final Nr. P4.064

Development of EAST spectroscopy diagnostics based on neutral beam injection

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Several spectroscopy diagnostics based on heating neutral beam injection (NBI) have been recently developed in EAST tokamak, i.e. the Charge eXchange Recombination Spectroscopy (CXRS), direct current Beam Emission Spectroscopy (DC-BES), alternating current Beam Emission Spectroscopy (AC-BES) and Fast-Ion D**α** (FIDA) diagnostics. These diagnostics provide tools for some important plasma parameters. The CXRS diagnostic system which has 25 toroidally located channels and 25 poloidally located channels, combined with a 10-channel DC-BES diagnostic, provides the space- and time-resolved profiles of impurity ion temperature, rotation as well as density. The 16×8-channel AC-BES system will provide a 1M temporal and a 1cm spatial resolution density diagnostic in a rectangular area in the cross section. The 46-channel FIDA system, which has a temporal resolution of less than 5 us and a spatial resolution of about 3cm, is used to provide the distribution of fast ion. All of these diagnostics are focused on the area from the core plasma to edge of the low field side. Physical plan in the next EAST experiment is presented, together with an idea of Ultrafast-CXRS diagnostic system in the nearly future. Acknowledgment *This work is supported by National Magnetic Confinement Fusion Science Program of China under Grant No. 2012BG101001.

Id 900

Abstract Final Nr. P4.065

Core Plasma Zonal Flows in HT-7 tokamak with Collective Scattering Density Fluctuation Measurement

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The zonal flows are intensely studied in recent decades for the importance in regulating turbulence and the key role in triggering L-H transition. Until now, there are mainly two approaches using the density fluctuations to search for the zonal flows indirectly. The first approach is measuring the plasma rotation velocity via the Doppler shift effect. Another approach is using the envelope analysis to deduce the zonal flows information embodied in interactions between the zonal flows and turbulence. For understanding the measurement of zonal flows using density fluctuations, the comparison between them should be discussed in detail. In this paper, we firstly extend the indirect approach using Instantaneous Frequency Method (IFM) on the density fluctuations measured by the CO2 laser collective scattering diagnostics in HT-7 tokamak to estimate the velocities of plasma poloidal rotation. The error analysis shows that the estimated poloidal rotation velocities are very reliable with high signal-to-noise ratio.A coherent mode is observed in the fluctuations of poloidal velocities with the mode frequency from 10kHz to 20kHz. It is identified as geodesic acoustic mode (GAM) zonal flow with poloidal symmetry (m=0) andits mode frequency coinciding with the theoretical expected GAM frequency, which is decided by the local plasma temperature. In the meantime, the envelope analysis is carried out on the high frequency density fluctuations.The relative amplitude of GAM in the envelope depends on the filter band of density fluctuations. In addition, the phase shift between the GAM radial electric field and the envelope of density fluctuations is proved to be radians. These results strongly recommended that the envelope modulation on the density fluctuation only reflects the shearing effect by the GAM. The results confirm that the envelope modulation in the high frequency density fluctuations only comes from the shearing by GAM.

Id 939

Abstract Final Nr. P4.066

Overview video diagnostics for W7-X stellarator

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The commissioning phase of the superconducting Wendelstein 7-X (W7-X) stellarator starts in spring this year and a first plasma operation campaign is expected to begin one year later. One of the diagnostics needed from the first operational day is the overview video diagnostics based on 10 toroidally viewing ports. The 10 channel video diagnostic system covering the whole torus interior can be used not only to observe the plasma but also to detect irregular operational events which are dangerous for the stellarator itself and to send automatic warning for the machine safety system. The quite harsh (significant gamma, neutron and heat radiation, strong - several Tesla - magnetic field) stellarator environment raises the requirements for the components (vacuum window, optical elements and video cameras) of the video system. The stellarator experiment has a complicated geometry: the plasma radiation can be observed only through a 2m long, tight (diameter: 0.12m) and bended observation port. Additionally, the length of W7-X discharges will be extended up to half an hour resulting in a video data amount which cannot be stored in conventional fast framing camera heads. The demand for high speed real time processing of video camera data is increasing in hot magnetically confined plasma control applications and is especially important for such long discharges. To meet the above requirements, a new video diagnostics is designed and manufactured, including an actively cooled pinhole protected vacuum window, optics, and a new intelligent camera (EDICAM, able to perform real time event detection) both located on the plasma side of the tangential AEQ port. In this contribution the whole video diagnostics system is documented including all the developed pieces of hardware (port docking mechanism, optics , camera electronics) and software (control and data processing software as well) and key elements are discussed in details.

Id 402

Abstract Final Nr. P4.067

Development of neutral lithium injector for beam emission spectroscopy plasma diagnostic system

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The edge behaviour of magnetically confined fusion plasmas plays main role in the plasma performance. The mechanism of H-mode and the L-H transition is still not fully known but from point of view of the confinement this is one of the key issues. Because of these reasons the measurements of the transport and of different magneto-hydrodynamic phenomenas in the plasma edge are indispensable. The number of diagnostic systems which are capable to measure in this area with good temporal (1 µs) and sufficient spatial (1 cm) resolution is limited. Lithium beam based beam emission spectroscopy (BES) is one of these methods, and has been widely used for the measurement of electron density and its fluctuation in the edge region. The Lithium atoms are excited in the plasma and they emit a photon with characteristic wavelength. This light intensity can be measured and used indirectly for density reconstruction and directly for density fluctuation calculation. Although the Lithium beam based BES limited by the beam penetration into the edge region of fusion plasmas the reachable information are unique taking into account the spatial and the temporal resolution. The aim of this development is to evolve an experimental Lithium beam injector, which is able to reach 60keV beam energy to measure of the excited Li 2p-2s state. The result of this work is two similar injectors which have been developed and manufactured in parallel in the WIGNER Research Centre for Physics. One of the two systems was installed on K-STAR in 2013 and the installation of the second one is presently ongoing on EAST. The goal of this paper is to represent the design of the developed injector from mechanical point of view such as: ion source, recirculating ion neutralizer, beam manipulator and beam profile monitor system and a new magnetic shielding concept.

Id 869

Abstract Final Nr. P4.068

Firmware and software development for Event Detection Intelligent Camera

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An innovative fast camera (EDICAM – Event Detection Intelligent CAMera) was developed by MTA Wigner RCP in the last few years. This new concept was designed for intelligent event driven processing to be able to detect predefined events and track objects in the plasma. One of the most important advantages of this hardware is a 10G optical link which ensures very fast communication and data transfer between the PC and the camera, making built-in camera memory unnecessary. A new firmware and software package is under developement. It allows to detect predefined events in real time and therefore the camera is capable to change its own operation or to give warnings e.g. to the safety system of the experiment.

Id 822

Abstract Final Nr. P4.069

High Temperature Superconductor Current Leads for the Tokamak JT-60SA

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High Temperature Superconductor Current Leads (HTS •]CL) are nowadays the preferred choice for supplying large electrical currents from the power supply to a superconducting magnet system. Although such CLs are more expensive, the significant reduction of the power consumption at room temperature has a big impact on investment and operation costs of the cryogenic system. Therefore large superconducting magnet systems in operation (e.g. LHC at CERN) or under construction (W7 •]X, JT •]60SA, ITER) use HTS •]CL. In all these projects Bi •]2223 material is being used. After the manufacture and test of the 14 HTS •]CLs for the stellarator W7 •]X the Karlsruhe Institute of Technology (KIT) has agreed to construct and test all HTS •]CLs for the tokamak JT •] 60SA presently under construction in the frame of the Broader Approach agreement between Europe and Japan. In total 6 HTS •]CLs for 26 kA for the TF coils and 20 HTS •]CLs for 20 kA for the EF and CS coils will be provided. The design of these HTS •]CLs is similar to the design of the W7 •]X current leads and the ITER current lead demonstrator built by KIT. A resistive copper heat exchanger with a warm contact covers the temperature range from 300 K to 60 K and is cooled by 50 K helium. An HTS module is connected to this heat exchanger covers the range from 60 K to 4.5 K cooled by heat conduction. The cold side of the HTS module is connected to the superconducting feeder cooled at 4.5 K. It is equipped with an Nb3Sn insert to reduce resistive losses. The HTS •]CLs are designed to withstand voltages of up to 10 kV in Paschen conditions during a fast discharge of the coils. The paper will describe the design and manufacturing status of the HTS CL, and show first results of the tests.

Id 141

Abstract Final Nr. P4.070

Electromagnetic and mechanical analysis of a toroidal field coil winding pack for EU DEMO

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Studies of ambitious fusion demonstration reactor DEMO are underway in most parts of the world, so as in Europe, so called EU DEMO. Actual design parameters for EU DEMO are taken from the result of so-called PROCESS system code. Using these results as basis winding pack of toroidal field coil (TFC) has been proposed using high temperature superconductors (HTS). Here the REBCO superconductor is a promising candidate. From this the cable space area, , the conductor current, the number of turns in one TFC, the peak magnetic field at the conductor, coil inductance along with estimation of temperature margin and the hot spot temperature in case of a quench and subsequent safety discharge of the coil has been obtained [1]. With these results it has been demonstrated that at 4.5 K the actual available HTS conductor can be used to design a TFC for EU DEMO within available space. As an extension of the previous work, the results of the electromagnetic and mechanical design calculations of a TFC conductor and winding pack will be presented. The paper will address parameters such as Lorenz forces, and evaluate stress and strains in the stainless steel jacket and of the turn insulation. This simulation will help in estimating jacket thickness. [1] P. V Gade, C. Barth, C. Bayer, W. H. Fietz, F. Franza, R. Heller, K. Hesch, and K. Weiss, “Conceptual Design of a Toroidal Field Coil for a Fusion Power Plant Using High Temperature Superconductors,” vol. 24, no. 3, 2014.

Id 235

Abstract Final Nr. P4.071

Analysis of the flow imbalance in the KSTAR cryogenic circuit

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The fourteen Poloidal Field (PF) superconducting magnets of the Korea Superconducting Tokamak Advanced Research (KSTAR) are cooled by forced flow (pressure gradient: 2 bar) of the supercritical helium (SHe) with 4.5 K and 5.5 bar. The mass flow rate of SHe is required more than 300 g/s for PF magnet system, and it is circulated by the cold rotating circulator in the Helium Refrigerator System (HRS). The SHe is distributed properly to the each PF magnets at the Helium Distribution System (HDS), which contains five cryogenic valves to control the flow rate for the cooling of the fourteen PF magnets. It travels about 50 m through the transfer line including in-cryostat helium lines which have the difference of elevation between the upper and the lower magnet and the conductance of the each cooling channel might be different even in the same cooling channel length. It makes the imbalance of flow rate between upper and lower magnets, and the slightly temperature difference of the inlet and the outlet between them. These thermo-hydraulic conditions of the SHe could have influence on the operation of the superconducting magnet. Therefore, in this paper, themo-hydraulic behavior in the KSTAR cryogenic circuit, which is quasi closed system, will be investigated during the PF magnet operation.

Id 157

Abstract Final Nr. P4.072

Final Design of the Korean AC/DC Converters for the ITER Coil Power Supply System

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The final design of the ITER TF, CS, CC and VS AC/DC converters has been completed to implement ITER requirements following the detailed design and refinements of the preliminary design. The performance of the converter has been demostrated by anlysis. The converter design implements the electrical fault suppression capability under short circut conditions to maximize the availability . The system integrity is maintained from the mechanical stresses under seismic and electomagnetic loads. The real time control cycle of 1 kHz for the converter plants feeding magnetically coupled superconducting coils is implemented following ITER plant control deseign handbook. The number of parallel thyristors and the rating of fuses are coordinated to keep those devices within the explosion limit even under most severe fault conditions. The impedance of the converter transformer has been optimized taking into account the energization inrush current, short circuit current, reactive power consumption and the available DC voltage. To keep the system integrity, AC/DC converters are mechanically divided into transformers, AC busbars, 6-pulse bridges, DC interconnecting busbars and DC reactors, then all subsystems are decoupled by flexible links. To provide real time network communication down to the converters, a one GbE link is deployed between master controllers and local controllers. An IEEE1588-PTP is implemented to the embedded controllers for precision time synchronization. This paper describes the detail solutions implemented in the final design for ITER AC/DC converters with R&D results of converter prototypes. Also it shows test results of hardware interfaces and algorithms implemented in controllers, which is verified by a real time digital simulator test bed.

Id 507

Abstract Final Nr. P4.073

Development of MgB2 superconducting wires for the low activation superconducting magnet system operated around core D-T plasma

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MgB2 superconducting wire is one of the promising to apply for the low activation superconducting magnet system to confine D-T plasma, because the half-life time of the MgB2 is estimated to be about 1 month and is much shorter than that of Nb-based superconductors. In the advanced D-T fusion applications, we think that MgB2 wire is the alternative conductor of Nb-Ti wire for the correction coil and Poroidal field coil around D-T core plasma. However, the critical current density (Jc) property of MgB2 wire is still insufficient compared with Nb-Ti wires. In previous studies, we succeeded to fabricate the MgB2 wire using boron-11 (11B) isotope as the boron source. In the Mg(11B)2 matrix, the secondary phase (impurity and non-reactive) and voids were observed in the matrix after the heat treatment and then these are the lowering factors of Jc property. The primary particle sizes of the boron powder is one of the effective factors to increase MgB2 volume fraction, and we tried to fabricate the nano particle size boron powder below 1 μm. The nano particle sized boron powder was made by the Ar gas jet-milling with the cyclone classification. The various classified nano particle boron powders as the function of milling times were prepared. The workability of the in-situ Cu addition MgB2 multifilamentary wires was enhanced by the fine particle sized boron powder. Highly critical transition temperature (Tc) values of the samples using classified nano boron powder were obtained to about 37 K. The comparisons of microstructure and Jc-B property by the different particle size and impurity contamination due to the jet-milling and classification is also reported. This work was mainly supported by Grant-in-Aid for Scientific Research (C) 25420892 and NIFS Fusion Engineering Research project (UFFF036 and UFAA014) and the NIFS Collaboration Research program (KECA009 and KECF004)

Id 888

Abstract Final Nr. P4.074

Study on mitigating method of large disturbance for helium subcooling system of the LHD helical coils

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The cooling system for the helical coils of the Large Helical Device (LHD), which is an experimental device for fusion plasmas, was upgraded in order to improve the cryogenic stability of the coils by lowering the operating temperature in 2006. In the system, liquid helium is subcooled to 3.0 K in a heat exchanger of a saturated helium bath and is supplied to the coils at the nominal mass flow rate of 50 g/s. The bath pressure and temperature is reduced by a series of two centrifugal cold compressors with gas foil bearing. In the steady state subcooling operation, the temperature is stabilized within range of 0.01 K with automatic mass flow control of helium gas through the cold compressors by a heater in the bath. So far, the total time of subcooling operations with the upgraded system exceeded 15,000 hours. It is important to mitigate large disturbance in the cooling system to operate the system reliably and safely. In the present study, the thermal hydraulic behavior of the system was investigated after a fast discharge of the coils for coil protection. And also, an effect of the automatic helium flow control by the heater on the mitigation of the disturbance was analyzed. Consequently, it is found that the cold compressors could avoid the surge and keep working without any troubles thanks to the automatic helium flow control.

Id 923

Abstract Final Nr. P4.076

Magnet system of the tokamak T-15 upgrade

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At present the tokamak T-15 is upgraded. Magnet system of the tokamak T-15 upgrade will provide the obtaining and confinement of the hot plasma in divertor configuration. Plasma parameters are: major radius - 1.48 m, minor radius - 0.67 m, elongation - 1.7-1.9 and triangularity is 0.3-0.4. Magnet system includes the toroidal winding and poloidal magnet system. Toroidal winding consists of 16 D shape coils forming the arched structure. Toroidal magnetic field at plasma axis is 2 T. Level of ripples at outboard plasma boundary is about 0.8 %. Each coil contents 50 turns which wound by hollow conductor made of silver-copper alloy. Winding packages are placed in cases made of stainless steel (321) with low magnetization. Toroidal winding is charged by eight thyristor convertors (22 kA, 1 kV each). A current plateau duration is ten seconds at B0= 2T. The poloidal magnet system (PMS) is capable of realizing the divertor both with single null and double null magnetic configuration. PMS includes a central solenoid and six PF coils (PF1 – PF6). Central solenoid consists of three separated coils (151,449,151 turns). Each coil is wound by hollow conductor made of silver-copper alloy. Coils are charged by independent power supply systems. Magnetic flux swing in central solenoid IIIcs is about 6 Wb (Ics = ± 40 kA). Six PF coils are placed outside toroidal winding. All coils are wound by hollow copper conductor. For fast plasma position control the four frame form active control coils (10 copper turns in each) are placed around the torus in the space between vacuum chamber shell and toroidal winding. All elements of magnet system will be manufactured to the end of 2015. The tokamak T-15 upgrade should be introduced into operation in 2016.

Id 811

Abstract Final Nr. P4.077

First Application of High Temperature Superconducting (HTS) TF Coils on a Tokamak

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The ‘spherical tokamak’ (ST) has demonstrated record efficiency, and is valuable for physics studies. However neutron production and fusion power increase as a high power of toroidal field, and to date, STs have been limited to toroidal fields of (mostly) 0.5T or less, limiting their application as practical fusion devices . The low fields arise because to obtain a high field at the plasma major radius together with the small central column diameter means that the centre column part of the TF magnet would have to carry huge currents in very large magnetic fields. Using copper invokes prohibitive resistive losses; conventional superconductors are limited both by their critical current at the high fields required, and by the demands of cooling and support structures. However High Temperature Superconducting (HTS) materials, when operated at temperatures some way below their transition, can carry high current under high field, and hence have the potential to provide high current density in tokamak TF coils. Development of a compact tokamak is underway; the ST25-HTS device is a 0.25m major radius tokamak of toroidal field 0.1T fitted with a six-limb TF magnet and two PF coils, all made from HTS flat tape YBCO conductors. Cryogenic cooling to ~20K is provided by a combination of helium gas flow and mechanical (Gifford-McMahon) cooler. The ST25-HTS coils are a route to improving our understanding of HTS material in tokamak coils and will demonstrate many of the techniques required to construct high field compact tokamak HTS coils. Results of TF coil tests on ST25-HTS are presented and major learning and challenges are discussed. ST25-HTS will be the first tokamak to be made entirely using HTS coils, and is a valuable step in evaluating the engineering challenges for a spherical tokamak to provide a compact fusion energy power plant.

Id 623

Abstract Final Nr. P4.078

Numerical Analysis of Modified Central Solenoid Insert Design

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The United States ITER Project Office (USIPO) is responsible for fabrication of the Central Solenoid (CS) for ITER project. The ITER machine is currently under construction by seven parties in Cadarache, France. The CS Insert (CSI) project should provide a verification of the conductor performance in relevant conditions of temperature, field, currents and mechanical strain. The US IPO designed the CSI that will be tested at the Central Solenoid Model Coil (CSMC) Test Facility at JAEA, Naka. To validate the modified design three-dimensional numerical simulations were performed using coupled solver for simultaneous structural, thermal and electromagnetic analysis. Thermal and electromagnetic simulations supported structural calculations providing necessary loads and strains. According to current analysis design of the modified coil option 5 satisfies ITER magnet structural design criteria for the following conditions: 1) room temperature, no current, 2) temperature 4K, no current,3) temperature 4K, current 60 kA direct charge 4) temperature 4K, current 60 kA reverse charge. Fatigue life assessment analysis is performed for the alternating conditions of: temperature 4K, no current, and temperature 4K, current 45 kA direct charge Results of fatigue analysis show that parts of the coil assembly can be qualified for up to 1 million cycles Distributions of the current sharing temperature (TCS) in the superconductor wereobtained from numerical results using parameterization of the critical surface in the form similar to that proposed for ITER. Special ADPL scripts were developed for ANSYS allowing one-dimensional representation of TCS along the cable, as well as three dimensional fields of TCS in superconductor materia.

Id 1032

Abstract Final Nr. P4.079

Analysis on magnetic field power supply of HL-2M tokamak based on the pulsed operation of motor generator

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The control precision and stability of the magentic field is essential for the research of high-beta steady-state plasmas in tokamaks. The flywheel motor generator sets operated in pulse mode are the main power source of HL-2M tokamak. Concretely for the power supply of the magnetic field coils on HL-2M, it consists of synchronous generators with related transformers and converters. The energy is stored mechanically in the shafting of motor generator sets, and it is transferred to the toroidal field coils by firing the exciter of the synchronous generator. In this case, modeling and control of the six-phase synchronous generator with corresponding exciter is the one of key issues for the precise and stable control of the field current. The analytical model of the six phase generator with exciters is specifically developed from the dual DQ transformation. Based on the electromagnetic dynamics which describe the flux linkage changes in DQ frame, the six-phase generator is represented by the exciting voltage controlled current source. On the other hand, the electromagnetic torque is calculated from the interaction between the flux linkages and currents in DQ frame, this torque drives the shafting angular speed to decrease from an initial angular speed, meanwhile the energy is released. The dynamics of the toroidal field current scenario is simulated on the basis of this model built in DQ frame. The elementary results show the consistency with the discharge results of HL-2A, and the model is effective for the toroidal field current control. Furthermore, this model can be adopted for the preprogrammed feedback control of toroidal field current in HL-2M. Moreover, this model is also suited for the simulation and analysis of synchronous machine - converter systems.

Id 252

Abstract Final Nr. P4.080

Investigation of Series-Connected IGBTs in fast high-voltage circuit breaker

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Advanced tokamaks require facilities that are working in high voltage circumstances, such as power supplies for the coils and for the laser. Some devices are also required to operate at high frequency. These requirements make the IGBT an ideal choice as the basic component of switches, for it has self turn-off ability, high switching frequency, and convenient operation. The application of series connection of IGBTs could make devices operate in a higher voltage level than only one IGBT. However, due to individual parameter differences of each IGBT circuit, it is difficult to achieve a proper voltage balance among the IGBTs, and the huge difference of the unbalanced voltage could cause a failure of these devices. This paper presents an active gate control method to compensate the trigger time of each IGBT's gate signal, realizing flexible adjustment of the switching time so that the IGBTs can switch on and off simultaneously. In addition, considering the different distributed capacitance of individual IGBT, vertical-arranged structure of the IGBTs has been applied to decrease the distributed capacitance, and subsequently decrease its influence to voltage sharing of series-connected IGBTs. The test environment consists of a 1500V/10A circuit with 3 IGBTs which are connected in series. The experimental results shows that the method can effectively improve the dynamic-voltage balance and the IGBTs connected in series work stably. It also has significant reference value for practical engineering applications.

Id 481

Abstract Final Nr. P4.081

An ohmic field power supply based on a modified igbt h-bridge for sino-united spherical tokamak

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Air core solenoids are widely used in spherical tokamaks, since their central stacks are quite compact and difficult to deploy iron cores. However, much higher current are required to generate the same amount of flux in the solenoids with air cores than with iron cores. Therefore, the power supplies of ohmic fields in STs are more challenging than those in conventional tokamaks with similar size. In this paper, the power supply for the ohmic field coils in sino-united spherical tokamak (SUNIST) is described. This power supply utilizes a modified H-bridge circuit, in which the high current switches are made by ten insulated gate bipolar transistors (IGBTs, rating 3.3 kV/1.2 kA) connected in parallel. A high voltage capacitor bank (1.2 kV/60 mF) and a low voltage capacitor bank (600 V/ 500 mF) store the energy required in different discharging stages. An low inductance high power resistor (0.3 Ohm) made by stainless steel foils is used to help the circuit to generate a current with high derivative in the ohmic field coils, which are required for gas breakdown, plasma current initiation and ramp up. The long time small loop voltage required for maintaining the plasma current is realized by the low voltage capacitor bank. For each tokamak discharge, all the switches of this power supply only need to be switched on and off one times with an interval of several milliseconds. Such slow operations of the IGBTs lower the risk of failure and minimize the technical difficulties of dynamical current sharing. The power supply is low-cost, compact, and reliable. It has successfully worked for more than 10000 times in recent two years.

Id 898

Abstract Final Nr. P4.082

Optimal design considerations for a Tokamak with a toriodal field magnet system using high-temperature superconductors

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High temperature superconductors (HTS) have the potential to reduce the reliance on helium coolant and dramatically reduce the cost of electricity (COE) from a fusion power plant. The systems code PROCESS, developed at CCFE, allows scans of important tokamak design parameters to be undertaken to establish new optimum operating conditions for HTS toroidal field (TF) coils. The operating magnetic field and temperature for the TF coils and a range of properties for the component superconductor (including cost, critical current density and stress tolerance) have been investigated, whilst ensuring that all physical constraints are met (e.g. the maximum power load on the divertor). We report optimal designs for the TF coils in a 500 MW power plant that minimises the COE. Y1Ba2Cu3O7 (YBCO) HTS tape has been investigated, being the most promising HTS material for a fusion power plant. The self-field, 4.2 K engineering critical current density, JE, was allowed to vary from the present value of 4,144 A.mm-2 up to 41,440 A.mm-2 to anticipate future improvements in tape manufacturing techniques. Scaling laws were then used to model the strain, temperature and field dependence of JE. We confirm that at present it would be uneconomical to build a fusion power plant using YBCO. However, we find that a cost reduction of ~2 would make YBCO tape competitive with Nb3Sn, whilst an increase in the critical current density by a factor of 3 would make it possible to operate economically at temperatures as high as 40 K (using Neon as the coolant). There are only very small savings to be made by increasing JE further. Further to this, if new materials are developed which could increase the allowable stress on the TF coils by a factor of 2 (the steel is currently limited to a maximum stress of 667 MPa), the maximum field could be increased from its current limit of ~13.5 T up to ~17.5 T. YBCO could then offer up to a 10 % reduction on the COE, as compared with Nb3Sn.

Id 387

Abstract Final Nr. P4.083

The comparison of heat flux on lower divertor in KSTAR

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Heat flux measurement at divertor is one of critical issues in ITER. For the monitoring and estimation of the heat flux in Korea Superconducting Tokamak Advanced Research (KSTAR), arrays of 200 thermocouples (TC) were installed on Plasma Facing Components (PFCs) at different poloidal and toroidal locations around the torus, and record temperatures every 1 sec. Due to the systematic limitation such as show response time of TC (~ 1 sec) and large spatial distance between thermocouples, the heat flux on divertor was estimated by simple calculation concerning the tile volume and heat capacity as time and volume averaged value.The heat flux at divertor is higher than that at inboard limiter or passive stabilizer by a factor of 2. The heat flux at lower divertor is higher than that at upper divertor by 3-8 times. It seems that lower-single-null-like configuration was obtained, despite of the shape control was intended to have up-down symmetric D-shape. Further effects of other controls on the divertor heat flux were observed: When the in-vessel cryo-pump (IVCP) is operated to control of particle flux at divertor, the heat flux pattern has changed significantly. The highest heat flux region of central divertor (CD) and inboard divertor (ID) is moved to inside and heat flux of outer divertor (OD) is decreased by 2-3 times than that without IVCP. The resonant magnetic perturbation (RMP) is operated to mitigate or suppress of Edge Localized Mode (ELM). As a result, the heat flux on divertor is decreased from 35 to 26 kW/m2, especially that on CD is decreased by 2-4 times and that on OD is increased by 2-2.5 times than that without RMP.For the longest H-mode pulse of 22 sec shot, the heat flux of lower OD is 73 kW/m2, which is the maximum heat flux for a shot obtained in 2013 campaign. As reported in our previous publications, the input power increased through campaigns, the heat flux on divertor has been increased as well. The effects of various plasma controls on the heat flux reported in this paper give valuable information on the feedback control in combination with survey IR camera for machine safety.

Id 178

Abstract Final Nr. P4.084

Effect of wortex flow on the heat transfer coefficient for ITER blanket shield blocks

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The main function of blanket Sheild Block (SB), one of the in-vessel components, is to provide nuclear shielding to Vacuum Vessel (VV) and remove the heat load deposited on the SB. In conceptual design of SB, a poloidal cooling concept was adopted to consider the benefits such as its manufacturability improvement. After conceptual design review, the interfaces with blanket first wall or VV were changed drastically, therefore cooling channel in a SB was redesigned following the design protocol, relevant codes and standards. Since the preliminary concept design, the hybrid cooling concept, which consists of poloidal and radial, was adopted for considering interface changes. The design activities for optimization of cooling performance have been implemented by blanket integrated product team composed mainly of ITER organization and domestic agency. Heat Transfer Coefficient (HTC) is a important factor, which determines the cooling performance of SB. Also, it generally depends on the type and velocity of fluid. According to results from hydraulic analysis of SB, relatively low velocity regions were observed at radial cooling channels. The objective of this study is to verify the HTC in relatively low velocity regions for design feasibility of SB. In order to investigate the effect of velocity on the HTC, numberical analysis was implemented with simplified model. In this anlaysis results, HTC was lower than that from hydraulic analisys of SB. In order to clarify this reason, the charateristics of fluid for both models and their dependence on the HTC were investigated. The heat transfer was enhanced by vortex effect due to the improvement in the field synergy.

Id 207

Abstract Final Nr. P4.085

Manufacturing and high heat flux testing of tungsten brazed mock-ups for KSTAR divertor

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Development of tungsten brazing technology for the upgraded KSTAR divertor, in which the tile material needs to be bonded onto the heat-sink plate to exhaust high heat load in plasma operation, was launched in early 2013. ITER grade tungsten block was brazed on the CuCrZr alloy in vacuum pressure at a temperature of 980 °C for 30 minute using silver free brazing alloy. A OFHC-copper was used as an interlayer between tunnsten and the CuCuZr because of its low yield strength and low elastic modulus. The brazing alloy is a 0.05 mm thickness of plate of which component is a Ni-Cu-Mn. It is found that the optimal surface roughness on tungsten is about 6 µm Rs and the optimal loading on mock-up is about 200 g/cm2 . Ultrasonic test, shear strength test, tensile strength test, bend strength test, and scanning electron microscopy were carried out to check the joint condition and strength between tungsten and substrate material. Tungsten brazed mock-ups with a cooling tube were tested at an electron beam facility, koHLT-EB(Korea Heat Load Test Facilities-Electron Beam) in KAERI(Korea Atomic Energy Research Institute). The high heat flux test was performed for tungsten brazed mock-ups under heat flux of 5 MW/m2 with more than 1,000 cycles, which had been chosen by several requests from the thermo-hydraulic analysis results with ANSYS code. All the tungsten brazed mock-ups met the requirements (heat flux test up to 5 MW/m2 and 1,000 cycles), and there was no delamination or failure at the bonding joints. This paper explains the manufacturing process of tungsten brazed mock-ups in detail and summarizes the results and process of the high heat flux test.

Id 812

Abstract Final Nr. P4.086

Pre-conceptual Design Study on K-DEMO Divertor System

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Based on the Fusion Energy Development Promotion Law (FEDPL) enacted in 2007 in Korea, a pre-conceptual design study for the Korean fusion demonstration tokamak reactor (K-DEMO) has been initiated with the uniqueness of high magnetic field (BT0 = 7.4 T), major and minor radii of 6.8 m and 2.1 m, and steady-state operation. To be ready for the start of conceptual design activity in 2015, design concept of major systems of K-DEMO needs to be prepared by end-2014 and the results of pre-conceptual design study on divertor system are summarized in this paper. K-DEMO tokamak plasma will be operated in up-down symmetric double-null configuration with high elongation (k=2) and triangularity (δ=0.63). Divertor first walls are configured to maximize the breeding area of blanket and to tilt the divertor targets at 10º against separatrix fieldlines to accommodate the conceived engineering limit of 10 MW/m2 of peak heatflux with ~40% core radiation limited by blanket first wall cooling capability and ~90% of divertor radiation at λq = ~1.5 mm [1]. Pressurized water is used as coolant for K-DEMO to be compatible with more efficient commercial steam generator technology. Divertor structure is composed of a tungsten-based target, steel-based structures carrying water coolant, a vanadium interlayer between them to enhance bonding, and supports. The developed pre-conceptual design concept on the target tilting, detailed material selection, cooling channel configuration, and robustness of supports, is supported by preliminary thermo-hydraulic and structural analyses using ANSYS. For vertical maintenance of K-DEMO in-vessel components, upper and lower divertor modules are subdivided into 32 toroidal modules, respectively. Also, to achieve a global tritium breeding ratio > 1.05, breeding blanket modules are partially placed around divertor structures. [1] T. Eich et al., Physical Review Letters 107, 215001 (2011)

Id 206

Abstract Final Nr. P4.087

The structure of the tungsten coatings deposited by Combined Magnetron Sputtering and Ion Implantation for nuclear fusion applications

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Combined Magnetron Sputtering and Ion Implantation (CMSII) technology was used for W coating of about 1800 CFC tiles for ITER-like Wall at JET and more than 1300 fine grain graphite tiles for ASDEX Upgrade tokamak. The limits of these coatings under high heat fluxes (HHF) were investigated as well. In this paper the structure of the W coatings with a Mo interlayer was examined in detail using SEM/FIB techniques before and after HHF tests. For deposited coatings two fine networks of nano-pores (10-20 nm in diameter) were detected for zones of 250-350 nm at CFC-Mo and Mo-W interfaces. During the HHF tests the dimensions and the number of the pores increased significantly. Besides the thermal fatigue and carbidization processes the presence of these pores seems to affect the limits of the W coatings. It was found out that the nano-pores networks were associated with the energy of the ions striking the coating during the deposition process. By optimizing this energy the nano-pores disappeared. The structure of the W coatings was also investigated by STEM and XRD techniques. The W grains for as deposited coatings have about 100 - 200 nm in width and up to 2 microns in length (perpendicular to coated surface). After HHF test at a peak temperature of 1350 C (500 pulses of 50 s pulse duration) the Mo coating was transformed into Mo_2C and a layer of about 1.5-2 microns of WC and W_2C was produced above the Mo_2C. The grain size in the WC is about 1 microns. The HHF tests carried out with an electron beam high temperature test facility (HHTF) demonstrated that the thermo-mechanical properties of the W coatings deposited by the new modified technology (CMSII-M) have been improved by a factor of about two.

Id 823

Abstract Final Nr. P4.088

Design of the Wendelstein 7-X Inertially Cooled Test Divertor Unit Scraper Element

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The divertor of the Wendelstein 7-X stellarator will evolve through three distinct phases of operation. For the first plasma, scheduled to begin in 2015, the device will operate without a divertor. For the next operational phase, scheduled to begin in 2016, an inertially cooled divertor (called the Test Divertor Unit, or TDU) will be installed so that the device can operate at higher power for short pulses. Finally, an actively cooled divertor will be in place for steady-state, full power operation, scheduled for 2019. Computational studies have indicated that for certain new plasma configurations in the steady-state operational phase, the ends of the divertor targets may receive heat fluxes beyond their qualified technological limit. To address this issue, a high heat-flux "scraper element" (HHF-SE) has been designed that can protect the sensitive divertor target region. The HHF-SE has been designed using carbon fiber reinforced carbon composite (CFC) monoblock technology which has been shown to withstand steady-state heat fluxes up to 20 MW/m2. The surface profile of the HHF-SE has been carefully designed to meet challenging engineering requirements and severe special limitations through an iterative process involving physics simulations, engineering analysis, and computer aided design rendering. The desire to examine how the scraper element interacts with the plasma, both in terms of how it protects the divertor, and how it affects the neutral pumping efficiency, has led to the consideration of installing an inertially cooled, but otherwise functionally equivalent version during the short pulse phase. This Test Divertor Unit scraper element (TDU-SE) would replicate the surface profile of the HHF-SE, but would be inertially cooled. The design and instrumentation of this component must be completed carefully in order to satisfy the requirements of the machine operation, as well as to interpret how the HHF-SE will perform during steady-state operation.

Id 278

Abstract Final Nr. P4.089

Amorphous tungsten oxide layers exposed to Magnum-PSI divertor-like plasma: retention, morphology and structural properties

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In the presented work amorphous tungsten oxide layers have been deposited on bulk tungsten (W) and on W coatings. Layer properties (composition, crystallinity and morphology) and deuterium (D) retention have been assessed before and after exposure to divertor-like plasma. Thanks to its physical properties (low erosion yield and high melting temperature), W has been selected for use in the divertor of ITER. Due to the strong chemical affinity of W with oxygen an W-oxide layer naturally exists on the metal surface. The presence of a W-oxide layer considerably lowers the W sputtering threshold energy. Modification and retention properties of thermally grown crystalline W oxide exposed to high energy (10 keV) D ions [1] and low energy (80 eV), low flux D plasmas (10^22 D/m^2*s) have been deeply investigated. These works show that D retention in W-oxide can be higher compared with bulk W[2]. With the purpose of better understanding the role played by W oxide in a real tokamak environment, we deposited amorphous W oxide layers (50-1000 nm), simulating different oxidation scenarios, both on bulk W and on compact nanocrystalline W coatings. The depositions were done by Pulsed Laser Depostion. The samples have been exposed to high flux divertor-like D plasmas (>10^24 /m2*s) in Magnum-PSI at various temperatures (270-640°C). Film features have been investigated using SEM, XRD, EDXS and Raman spectroscopy prior and after exposure. D retention has been characterized using TDS. The exposed W-oxide layers lose oxygen depending on temperature and D ion flux. At low temperature W-oxide exhibit lower D retention, ten times less, compared with W coatings. In all the exposed samples the oxide layer is not delaminated but micrometric cracks and nanometric cavities have been detected. [1] S.Nagata et al., Phys. Scr. T94 (2001) 106 [2] V.Kh.Alimov et al., J. Nucl. Mater. 409 (2011) 27-32

Id 552

Abstract Final Nr. P4.090

Engineering Challenges and Design Progress of the ITER Diagnostic First Wall

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Protecting ITER components from heat and neutrons while allowing viewing access for diagnostic performance leads to a conflicting set of requirements, and results in engineering challenges in the design of the Diagnostic First Wall (DFW) components. Most of the ITER components are shielded and protected by blankets mounted on the vacuum vessel walls. But at the upper and equatorial ports, where the vast majority of the diagnostics are positioned, port mounted blankets (namely DFW's) exist. The protection ability of these DFW's is compromised with an often-elaborate arrangement of apertures needed for viewing the plasma. Consideration to standardize the apertures for the many diagnostic systems, to limit DFW variations, was considered unfeasible, since standardized apertures could not provide the viewing angles, size and shapes of the apertures needed. Therefore, the DFW design concept needs to be adapted to a very wide range of configurations. Each DFW is a unique heat exchanger exposed to a plasma radiant heat load of 35w/cm2 and peak neutron fluxes of 8w/ cm3 (equatorial DFW) and 5w/ cm3 (upper DFW). Analysis indicated a bare stainless steel plasma facing surface, if recessed 10cm from the last closed flux surface, would sufficiently limit plasma impurities and could minimize the fabrication costs. The steel FW simplifies the components construction and reduces cost, but limits the design performance due to the relatively poor thermal properties of stainless steel. Due to the large variation in apertures, the components range in mass between the Equatorial DFW's (615kg to 981kg) and Upper DFW's (911kg to 1170kg). A successful Preliminary Design Review of the DFW was conducted in Dec 2013, and the design progression is now continuing toward a Final Design Review planned for the Dec 2014. This paper will discuss design challenges and features for the ITER "port mounted blankets" known as the Upper and Equatorial Diagnostic First Walls (UDFW & EDFW respectively).

Id 64

Abstract Final Nr. P4.091

Analysis of the Wendelstein 7-X Inertially Cooled Test Divertor Unit Scraper Element

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Early implementation of divertor components for the Wendelstein 7-X stellarator will include an inertially cooled system of divertor elements called the Test Divertor Unit (TDU) . One part of this system is a scraper element that is intended to explore methods of mitigating heat flux on the main TDU elements. This system will be in place in 2017, after a run period that will involve no divertor, and will precede steady state operation with actively cooled divertors scheduled for 2019. The TDU scraper element is an experimental whose loading conditions are somewhat uncertain at this time. The pattern of heat flux may vary from currently predicted distributions and intensities. The design of the scraper element must accommodate this uncertainty. Originally the mechanical design was to be based on extensive studies for the monoblock-based design of an actively cooled scraper element. An obvious simplification is the elimination of the manifolding needed for the water cooling. During the Conceptual Design of the TDU scraper element, a number of options were investigated. Larger tiles could be considered. Larger tiles and assemblies of tiles allow better control of surface tolerances during manufacture, but the size and tile design are limited by the deformations and stresses due to thermal gradients. Initial design targets for the heat flux were similar to the actively cooled specification. The shorter pulse lengths of the initial phases of operation were expected to allow inertial absorption of the steady state heat flux of up to 15 MW/m^2 for a six second pulse. However, it was found that peak temperature, thermal stress and distortion limit the design heat flux to 8 MW/m^2. A relatively large graphite block is used and is cut to provide relief of the tensile stress just below the surface of the tiles. There is a trade between the relief of stresses near the tile surface and the stress at the root of the cut. Thermal deformation of the larger tile is managed by supporting the tile in the middle to produce a curvature away from the plasma. Thermal ratcheting of the tiles and supporting structures is managed with adequate cooldown times, thermal anchors, where allowed, and thermal shields to limit radiative heat flux. Analysis of some of the initial concepts is presented along with analysis of the components chosen for the conceptual design.

Id 760

Abstract Final Nr. P4.092

Elucidation of sputtering mechanism for carbon-tungsten mixed layer during hydrogen implantation

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Tungsten(W) is one of candidates for plasma facing materials in fusion reactors. It is thought that W will be exposed to energetic particle such as hydrogen isotopes and carbon during the plasma operation and, not only the retention of hydrogen isotopes but also the desorption of hydrogen isotopes as form of hydrocarbons will be enhanced by the energetic particle implantation. The understanding of sputtering dynamics on W surface during hydrogen isotope ion implantation is important for the establishment of the fuel cycle and desorption mechanism of hydrocarbons for carbon-tungsten mixed layer during hydrogen(H2+) implantation was studied. The 10 keV C+ implantation with an ion flux of 1.0 × 1017 C+ m-2 s-1 and ion fluence of 1.0 × 1021 C+ m-2 was performed for W samples to form W-C mixed layer. After the C+ implantation, the H2+ implantation was performed with an ion flux of 1.0 × 1018 H+ m-2 s-1 at 673K. The energy of H2+ was changed from 0.3 to 3.0 keV. In-situ measurement of sputtered particles was performed by a quadruple mass spectrometer. Major chemical form of hydrocarbons during H2+ implantation was controlled by ion energy and CH3 was the major species by the 3.0 keV H2+ implantation, although CH2 was the key species by H2+ implantation with the energy less than 3 keV. Most of ion energy was consumed near the surface around 5 nm depth during low energy H2+ implantation compared to higher energy implantation, leading to the formation of CH2. Therefore, it was considered that the sputtering process has two pathways, namely physical sputtering and chemical sputtering and it is affected by energy given near the surface atoms. These results indicated that the chemical form of sputtering particles was controlled by the amount of transfer energy by H2+ implantation near the surface region.

Id 522

Abstract Final Nr. P4.093

Preliminary design and analysis of HL-2M first wall and divertor

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A flexible copper conducting tokamak HL-2M is being designed and constructed in Southwestern Institute of Physics of China. The plasma major radius is 1.78 m and minor radius 0.65 m with 2.5 MA plasma current. HL-2M is designed to operate under advanced divertor configurations, such as snowflake and tripod divertor configurations. Base on the divertor configurations, a preliminary structural design and analysis for the first wall and divertor are presented. The experimental plans for the coming year include running plasmas using the inboard wall center post as a limiter, so the centerpost is covered by all-graphite armor tiles. The height of these tiles is minimized to maximize the plasma volume, and electromagnetic loads on these tiles are investigated carefully to avoid damage of the tile and the pollution of plasma. Module armor design is adopted to facilitate the installation and maintenance of each tile. For the advanced divertor design, many possible configurations are envisioned, and a moveable strike point is also planned. To achieve that goal, a very large area of the divertor plate should be able to suffer high heat flux. So graphite tiles or CFC with powerful water cooled copper plate will be used as the targets. Due to compact design of divertor, the distance between the armor surface and bottom vessel wall is very limited, so a challenging design of the water cooling loops is put forward. For the outboard vessel wall, five bumper limiters are designed to prevent the plasma from hitting the vessel wall. Electromagnetic force and thermal stress analysis are presented in this paper to validate some fundamental and critical parameters, and the demands from other groups, such as diagnostics and plasma heating are also incorporated into the design.

Id 254

Abstract Final Nr. P4.094

Influence of microstructure on the thermal shock performance of CVD tungsten coatings

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Chemical vapour deposition (CVD) is a feasible way to fabricate tungsten for plasma facing components. Tungsten coatings deposited by CVD have high density, high purity, good thermal conductivity and good thermal shock resistance. CVD tungsten mock-ups also show good thermal fatigue properties. Thick CVD tungsten grown by a fast deposition rate has a rough surface and large columnar crystal structure and most grains on the surface exhibit cone shapes. Microstructure such as surface morphologies, grain size and grain orientation could have big influence on the material’s response to the operational loading conditions occurring in a tokamak. In this paper, the CVD tungsten coatings with thickness 2mm were prepared at different deposition rates. Thermal shock analyses were performed using an electron beam to study the influence of the variation in microstructure on the thermal shock resistance of the CVD tungsten. Repetitive ELM like loads were applied at various temperatures between RT and 600 oC with a pulse duration of 1 ms and an absorbed power density of up to 1 GW/m2. The induced thermal shock cracks and surface modifications were analyzed by surface profilometer, light microscopy (LM) and scanning electron microscopy (SEM). Damage behaviors related to the microstructural properties and loading conditions were discussed and presented. The result indicates that the specific surface morphology of the material has a significant influence on the surface cracking threshold of the CVD tungsten and crack propagation. The CVD tungsten coatings with a polished surface show superior thermal shock resistance as compared to the as-deposited coatings with a rough surface.

Id 490

Abstract Final Nr. P4.095

Methods to reduce contamination of the ITER VIS/IR diagnostic system first mirror

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Mirrors are essential components for many diagnostic systems of tokamak fusion reactors. The first element of the optical system is a single crystalline Mo mirror, called “First Mirror” (FM). It directly faces plasma being placed relatively close to the confinement chamber and reflects light signals to remote optical equipment. The optical properties of the unprotected FM in ITER imaging diagnostics degrade significantly in a short period of time because of deposition of particles formed in plasma-wall interactions. The deposited materials include beryllium and tungsten as well as carbon and hydrocarbons. For processes on a mirror surface, predicting the deposited layer thickness is not always possible, because the conditions are not fully known. Cleaning the mirror surface with for instance a hydrogen plasma is considered resulting in ion fluxes toward the FM surface. However, regular cleaning of the highly reflecting surfaces with relatively high ion energies, reaching hundreds of eVs, might result in reflecting properties degradation. It is not clear if contaminants like beryllium and tungsten could be efficiently removed with little FM surface damage. The number of cleaning cycles is also limited due to the nature of the ITER reactor performance. We propose a contamination prevention strategy based on reducing energy and intensity of the incident ion and atomic fluxes. This includes thermalisation of incoming ion species and fast neutrals by differential pressures and preventing wall sputtering with baffles. Influence of externally applied electrical field as well as the tokamak magnetic field on the ion species near the FM is investigated. Recommendations are also discussed on how to engineer fluxes inside the mirror chamber and how to combine geometry and pressures to reduce FM contamination. We expect that the proposed methods in combination with plasma cleaning will lead to substantial increase of FM lifetime.

Id 878

Abstract Final Nr. P4.096

Study of closed-divertor on liner divertor simulator TPD-Sheet IV

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In the design of fusion reactors for high power and long pulse operation, vast heat flux (> 10[MW/m2]) is expected to flow onto divertor target plates. In order to reduce this heat load, the divertor design on stable detached plasma formation must be realized. Experimental simulation on the V-shaped and long-leg target for closed divertor has been conducted in a linear divertor plasma simulator TPD-SheetIV. [1] Recently closed divertor concept such as V-shaped divertor or long leg divertor has been proposed and its optimization needs detail information on charged and neutral particles and their physical process. In order to understand the basic mechanism of detached plasma, many kinds of ion must be detected separately. [2] In our experiment, the hydrogen atomic and molecular ion currents (H+, H2+, H3+) are measured by omegatron type mass analyzer located behind target. The ionization and recombination events are discussed using the collisional-radiative (CR) model. When long duct is connected to V-shaped divertor, signal of H3+ is increase even in low gas puff case. This means the evidence of detactment plasma with little neutral back flow. [1] S. Tanaka, et al.,Fusion Sci. Tech. 63 (2013) 420 [2] A.Tonegawa, et al., Jpn J. Appl. Phys. 45 (2006) 8208

Id 919

Abstract Final Nr. P4.097

Raman investigation of irradiated amorphous carbon-tungsten materials

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Carbon based materials are one of proposed materials for being used in fusion devices, due to their good thermal conductivity. During the plasma wall interactions, the erosion, redeposition on the plasma facing surfaces and tritium accumulation in the divertor areas could take place. Covering the carbon-based materials with tungsten films increases the overall performances and decreases the formation of redeposited layer. Nevertheless erosion, cracks and formation of melt layers, could also affect the tungsten layer while plasma-wall interaction may still lead to dust and flake formation and depositions on the divertor surfaces, forming carbon/tungsten/carbon layered structures. In the present work we investigate the behaviour of such deposited carbon layer on the surface of tungsten under high power irradiation conditions. The structural modifications and carbon-tungsten interactions in the irradiated areas were investigated by micro-Raman spectroscopy using Ar ion laser with wavelength 514.5 nm. A graphite target covered with a 200 nm tungsten layer and a 1.5 µm amorphous carbon layer was irradiated by a 100 fs laser beam at 800 nm wavelength with 120 mJ energy per pulse. From the first laser pulse deposited carbon layer was already removed and more complex processes like tungsten layer ablation and amorphous layer graphitization take place for further pulses. The areas irradiated with fluencies below the ablation threshold, corresponding to the beam edges, present structural modifications and significant changes in the sp2/sp3 ratio. ACKNOWLEDGEMENTS C.P.L and A.M. acknowledge funding by the Romanian National Authority for Scientific Research, UEFISCDI, through project number PN-II-ID-PCE-2011-3-0522.

Id 978

Abstract Final Nr. P4.099

Fatigue testing of plasma facing materials for brazed divertor components

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Proposed high heat flux plasma facing component solutions for fusion reactors rely heavily on brazed two part structures often of tungsten armourer joined to EUROFER 97 steel in the case of helium cooled designs or potentially copper alloys if utilising water cooling. Due to differences in key physical properties, discontinuity stresses will under loading arise at the joint interface which is problematic when trying to ensuring component integrity during operational life. Loading will be cyclic in nature so fatigue is a potential failure mechanism and so conventional mechanical testing has been used to study this behaviour. Firstly EUROFER 97 steel and WL10 tungsten alloy have been tested by rotating bending to generate the first known S-N date for these materials by this test method. Notably the brittle tungsten alloy showed far greater scatter at higher stress levels but also longer fatigue life then the EUROFER 97. Though data points where limited, in the case of EUFOER 97 they could be used to start to estimate an S-N curve. A further set of axial tensile fatigue tests were carried out on two part dissimilar material brazed tungsten – copper specimens, with Orobraze 910 filler. Metrology assessment showed that there was a variable degree of both angular and axial misalignment after the brazing process. In testing, the ‘as joined’ specimens showed greater scatter and longer fatigue lives compared to specimens that had been machined post braze to remove local misalignment. Comments are made on the limitations and advantages of using mechanical testing as means of exploring the fatigue behaviour of divertor materials.

Id 1013

Abstract Final Nr. P4.101

Investigation of various nozzles configurations with respect to IFMIF and liquid walls concepts.

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The study of free surface flows is of great interest in fusion research, e.g. for IFMIF or the liquid walls concepts. On one hand, in the IFMIF project the main goal is to test candidate metallic materials in irradiation conditions similar to those found in a fusion reactor. More specifically, an intense neutron source will be produced by bombarding a liquid lithium target jet with two deuterium beams of 40 MeV [1]. The source will then be used to test samples of the candidate materials. On the other hand, the so called 'Liquid walls' project is related to the use of liquid film free surface flows as plasma facing components (PFCs) as an alternative to metallic plasma facing materials. Such method could result in important advantages, i.e. minimizing of corrosion defects and faster maintenance [2]. In both concepts the feeding of the liquid film is envisioned to be provided by a nozzle. The retaining of the flow stability after the nozzle is of significant research interest [1,2]. According to the literature, nozzles with profiles based on Shima's formulation favors such stability [1]. In that context, we tested several single and double reducer nozzle modules, of such type, by simulating a lithium/argon flow similar to the one expected in the IFMIF lithium target assembly. The results show that the use of a single reducer nozzle favours stability, and also that the efficiency of the double reducer nozzles depends on their reducing ratio. References 1) S. Gordeev, V. Heinzel & R. Stieglitz, Hydraulic numerical analyses of the IFMIF target performance, Fusion Eng. Des., 86 (2011), 2545-2548. 2) R.B. Gomes et al., Interaction of a liquid gallium jet with the tokamak ISTTOK edge plasma, Fusion Eng. Des., 83 (2008), 102-111.

Id 695

Abstract Final Nr. P4.102

The preliminary thermal hydraulic analysis of CFTER divertor

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China Fusion Engineering Test Reactor (CFETR) is proposed and being designed aiming at 50–200 MW fusion power, 30–50% duty time factor, and tritium self-sustained. One of the most challenging is the divertor design of CFETR. The thermal-hydraulic design of the divertor is particularly demanding because of the high heat loads on the plasma-facing components (PFCs) and the pressure drop related to the pumping power and the critical heat flux (CHF). In order to verify the reasonability of the divertor cooling structure, finite element (FE) calculations by ANSYS code has been carried out focusing on temperature distribution of the divertor and pressure drop of the cooling water. The velocity of the cooling water should be as high as possible concerning the heat load bearing of the divertor. But simultaneously the pumping power for the cooling system would be higher. In the paper, an optimal velocity is given as a compromise between the heat load bearing and the pumping power based on FE analysis. The cooling water circuit was also optimized for the purpose of reducing the pressure drop.

Id 476

Abstract Final Nr. P4.103

Study of the flow and heat transfer of supercritical water inside the first wall of water cooled solid breeder blankets

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Water Cooled Solid Breeder Blanket (WCSB) is one of the three major choices for Chinese Fusion Engineering Test Reactor (CFETR). To make CFETR more attractive, higher heat efficiency of more than 40% and a compact structure of plant system can be achieved by cooling the blanket with supercritical water at pressure of 25 MPa. But in the vicinity of critical point, strong variations of water properties combined with high flux can lead to deteriorated heat transfer, which consequently causes a severe increase of the wall temperature. In general, the empirical heat transfer correlations are not capable of predicting the heat transfer which has strong property variations, especially in case of deterioration. Hence in this paper the heat transfer behavior of the supercritical water and the mechanism of Heat Transfer Deterioration (HTD) inside the first wall will be intensively studied by means of Computational Fluid Dynamics (CFD). The effects of mass flowrate, heat flux, buoyancy, flow direction, etc on the behavior of turbulent flow and heat transfer of supercritical water will be detailedly analyzed. Based on these results, an optimized design solution for supercritical WCSB will be suggested.

Id 903

Abstract Final Nr. P4.104

Manufacturing Design of the ITER Vacuum Vessel Lower Port in Korea

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ITER vacuum vessel (VV) ports consist of upper, equatorial and lower ports. Korea Domestic Agency (KODA) is responsible for procuring port components at the equatorial and lower level including the local penetrations, VV gravity supports, neutral beam duct liners and sealing flanges. The main lower port structure can be divided into Lower Port Stub Extension (LPSE) and Lower Port Extension (LPE). The contract for the main port fabrication was signed with Hyundai Heavy Industries Co., LTD (HHI) on January 2010. After contract, manufacturing design has been developed by KODA and HHI to manufacture real product. As the first step of development of the manufacturing design, fabrication feasibility studies were carried out in accordance with the RCC-MR 2007. A full scale mock-up of the LPSE has been fabricated in order to verify weldability, applicability of Non Destructive Examination (NDE) and welding distortion. Fabrication sequence and manufacturing techniques have been established based on the results of a mock-up fabrication. Several qualifications were also conducted to confirm that the manufacturing design satisfies the requirements for bending, solution heat treatment, welding and NDE. 3D multi-part model was developed based on reference multi-body model and manufacturing drawings have been produced. Engineering analyses have been performed to guarantee structural integrity of manufacturing design and to minimize welding distortion. Through these activities, manufacturing design has been finalized and Manufacturing and Inspection Plans (MIP) have been prepared. Thanks to a long preparation activities since a contract with the supplier, manufacture of real product for LPSE was started at the end of 2012. In this paper, major technical results of manufacturing design will be presented for LPSE and LPE. Manufacture progress of real product for LPSE also will be introduced.

Id 282

Abstract Final Nr. P4.105

Piping Structural Design for the ITER Thermal Shield Manifold

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ITER is international fusion research reactor that aims to demonstrate possibility to produce commercial energy from fusion. Thermal shield (TS) is 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. The TS is actively cooled by 80K pressurized helium gas. The helium coolant flows from the cold valve box to the cooling tubes on the TS panels via manifold piping. Thermal contraction is one of the major loads for the manifold piping because it is cooled down from room temperature to cryogenic temperature (80K). The manifold support should absorb the pipe displacement due to thermal contraction as well as provide sufficient integrity for small deflection. This paper describes the manifold piping design and analysis for the ITER thermal shield. In order to accommodate the thermal contraction in the manifold feeder, a contraction loop is designed and applied. Guided cantilever beam method is used to determine the initial dimensions of the contraction loop to ensure adequate flexibility of manifold pipe. Global structural behavior of the manifold and support are investigated by a finite element analysis based on ASME B31.3 code. Several load conditions, such as dead weight, thermal contraction, seismic load, electromagnetic load and load combinations are considered in the global analysis. Maximum temperature difference between the active and the redundant pipes is also considered in the analysis for the conservative design. Detailed design of the support is described and its structural reliability is evaluated by a local finite element model.

Id 470

Abstract Final Nr. P4.106

Fabrication of a full-size mock-up for 10º section of ITER vacuum vessel thermal shield

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ITER Vacuum Vessel Thermal Shield (VVTS) is placed in a narrow gap between the Vacuum Vessel (VV) and the Toroidal Field Coil (TFC) to protect the TFC structure from excessive thermal radiation from the VV. The main material of VVTS is SS304LN. The overall height of VVTS is about 12 m and its panel thickness is 20 mm. The VVTS segment is a slender open structure and thus vulnerable to deformation caused by the manufacturing methods such as cutting, forming, bending and welding. Due to narrow space where VVTS is located, its tolerance requirement is so tight that the precise manufacture is indispensable. Therefore, full-scale mock-up test is required to validate manufacturing processes to avoid possible risks before manufacturing. In this paper, mock-ups of the VVTS 10º inboard and outboard section are described and test results are reported. All the manufacturing processes except for silver coating were tested and verified in the fabrication of mock-up. For the forming and the welding, pre-qualification tests were conducted to find proper process conditions. Welding sequence between flanges and shell was determined to comply with the ASME VIII Div.2. Welding was validated by non-destructive examination. After shell welding, shell distortion was measured and adjusted. Shell thickness change was also measured after bending, forming and buffing processes. Specially, VVTS ports need large bending and complex welding of shell and flange. Bending method for the complex and long cooling tube layout especially for the VVTS ports was also developed in detail. Bisectional flange joint was successfully assembled by inserting pins and tightening with bolt/nut. Bolt hole margin of 2 mm for sector flange was revealed to be sufficient by successful sector assembly of upper and lower parts of mock-up. Dimensional inspection of the fabricated mock-up was performed with a 3D laser scanner.

Id 475

Abstract Final Nr. P4.107

Development of the ITER Upper Port Stub Extension Assembly Tool and its Mock-up

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The ITER upper port stub extension (UPSE) of the vacuum vessel (VV) is classified into 2 categories according to its location and time to install; One is the central UPSE which is welded at upper central position of the VV sector and delivered after attached at factory. The other is the lateral UPSE which is welded at upper lateral position of the VV sector and the significant difference with the central UPSE is that the lateral UPSE is welded after whole VVs welded because this should be installed between each VVs. Eventually, the lateral UPSE should be handled and welded under the in-pit environment by the dedicated assembly tooling due to its assembly environment and procedure. According to IO's UPSE assembly scheme and procedure, the UPSE should be handled and installed by the dedicated assembly tool considering very limited space and crane accessibility conditions. When the UPSE is approached to the port stub of the 2 VV sectors at in-pit, limited space should be considered because there is already an upper port-shroud assembled at the assembly building during sector sub-assembly. Also, the UPSE should be manipulated accurately due to its very complicated procedure, assembly clearance and to satisfy welding preparation. The design of the dedicated assembly tool for the UPSE assembly has been developed by the Korean domestic agency (KODA). Adjustment system of this assembly tool has been also designed to meet the functional requirements requested by IO. For design verification of the UPSE assembly tool mentioned above, a mock-up has been fabricated in full size and tested according to the UPSE functional requirements for the assembly procedure. And the structural analysis results of the current design will be presented also.

Id 69

Abstract Final Nr. P4.108

Recovery process of wall condition in KSTAR vacuum vessel after temporal machine-vent for repair

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Efforts have been made to obtain high-quality vacuum condition that is essentially needed for the plasma experiments. During the past six Korea Superconducting Tokamak Advanced Research (KSTAR) campaigns, the baking and the GDC systems have played this role of achieving acceptable initial vacuum condition. Baking of the vacuum vessel, PFCs, and the pumping duct are performed solely or together with D2/He GDCs according to the initial wall conditioning procedures for the KSTAR device. This wall condition is generally conducted until impurities such as H2O, carbon and oxygen from the wall are reduced to below predefined levels. This condition is normally maintained before the KSTAR plasma experiment. However several times in the middle of several campaigns, the vacuum vessel should be vented to repair inside components damaged by high energy plasma, which are Diagnostic shutter repair, PFC repair, Thomson calibration, and water leak from NBI chamber, and so on. As the result of the vacuum venting, impurities in the vacuum vessel deteriorated again. For the quick restart of the campaign, a recovery process was proposed to make the vacuum condition acceptable for the plasma experiment in a far shorter time period than the initial wall conditioning procedures. After the recovery process, the following plasma experiment was well performed not making any plasma issues. In this paper, we present the recovery technique of the vacuum vessel condition after venting the vacuum vessel, and show some experimental results to support its validity.

Id 664

Abstract Final Nr. P4.109

Novel divertor design to mitigate neutron irradiation in the helical reactor FFHR-d1A

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The divertor heat flux in an LHD-type fusion reactor is considered to have a high peak of >10 MW/m2. Thus, a high-heat removal performance divertor needs to be developed. In the helical reactor FFHR-d1A, we are investigating the feasibility of employing a copper alloy for divertor cooling pipes because the neutron load on the divertor can be reduced by setting a blanket in front that prevents direct neutron irradiation to the divertor. The displacement per atom (dpa) at the inboard side of the torus, where the neutron irradiation is relatively high, is approximately 1.6 dpa/yr. The copper alloy limit is supposedly below 1 dpa/yr. In this case, the divertor at the region has to be replaced every seven months. The frequency of parts replacement affects the operation time and maintenance of a fusion reactor. If the position of the divertor is moved to a lower irradiation level area, the lifetime of the divertor component materials can increase. However, the vacuum vessel is limited by the coil support structure, which consists of the coil case, shell arm, and torus-shaped shell. The shell arm connects the coil case to the torus-shaped shell. To create a low neutron irradiation environment for the divertor, the shell arm has to be partially removed to allow the movement of divertor parts. Another advantage of this modification is that the divertor can be accessed from the upper side of the vacuum vessel, where the maintenance working method is easier than that from the inner side port. Analysis of the divertor leg trace, the neutron irradiation distribution, and the stress distribution in the support structure with partially removed shell arm was performed. The irradiation damage decreased to 0.1 dpa/yr, and the lifetime of the copper alloy in the divertor was estimated to be ten years.

Id 202

Abstract Final Nr. P4.110

Fracture Mechanics Analysis Approach to Assess Structural Integrity of the First Confinement Boundaries in ITER Generic Upper Port Plug Structure (GUPP)

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ITER diagnostics port plugs are plate structures placed into the ITER VV port extensions fixed through the ports flanges. They constitute common containers for a variety of diagnostics and shielding material. Generic upper port plugs (GUPP) are trapezoidal box structures around 5.5x1x.0.7 m and 15 ton that are part of the first containment barrier. They are classified as SIC-1 with regard to safety and QC-1 in terms of quality. A critical requirement of SIC-1/QC-1 structures is the need of an in-service inspections program (ISI). Following the principles of the Nuclear Basic Installations French Order of February 2012, the ISI program should be justified or alternatively, compensatory measures must be proposed to ensure that the equipment can provide its safety function with the required level of reliability. This analysis demonstrates structural integrity of the first confinement boundary in GUPP structures against cracking during service. This constitutes part of the justification to demonstrate that the non-aggression to the confinement barrier requirement may be compatible with the absent of a specific ISI program in the trapezoidal section. Since the component will be subjected to 100% volumetric inspections it can be assumed that no defects will be present before its commissioning. Cracks during service would be associated to defects under Code acceptance limit. This limit can be reasonably taken as 2 mm. Using elasto-plastic fracture mechanics (EPFM) an initial defect is postulated at the worst location in terms of probability and impact on the confinement boundary. Its evolution is simulated through FEA and final dimension at the end of service is estimated. Applying the procedures in RCC-MR 2007 (App-16) the stability of the crack is assessed. As relative high safety margin was achieved, a complementary assessment postulating an initial defect of 6 mm was also conducted. New margin calculated provides a more robust design.

Id 464

Abstract Final Nr. P4.111

3D Thermal-Hydraulic Analysis of two Irregular Field Joints for the ITER Vacuum Vessel

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The ITER vacuum vessel (VV) is located inside the cryostat and houses the in-vessel components. It is a double wall structure surrounding the plasma, where the volume between the external and internal shells is designed to allow the forced circulation of the cooling water through a very complicated structure of borated In-Wall Shielding (IWS) that works as neutron shield. The VV is made of 9 sectors, connected through splice plates to form the full torus. The regions at the interface between adjacent sectors are the so-called field joints. While each sector has its own cooling loop to remove the heat deposited by nuclear heating, each field joint is cooled by a separate circuit. Following the study [1] of a so-called "regular" field joint (RFJ), connecting two regular VV sectors, we present here the results of the ANSYS-FLUENT® 3D thermal-hydraulic analysis of two different irregular field joints (IFJs): an IFJ of type B, connecting the "irregular" VV sectors # 2 and # 3, characterized by an oblique equatorial port stub for neutral beam injection, to be simultaneously cooled with the IFJ, and an IFJ of type C, connecting the "irregular" VV sectors # 3 and # 4, characterized by the absence of equatorial port, both subject to a steady-state nuclear heat load. It is shown that the cooling capability of the sub-cooled water, entering the IFJs at 100oC and ~ 0.8 MPa, is sufficient to avoid hot spots above the design limits, while the pressure drop remains acceptable. The total number of cells needed to guarantee an accurate numerical solution is of ~ 40 million, including both solid and fluid components. The thermal-hydraulic performance of both IFJs turns out to be comparable with that of the RFJ. [1] L. Savoldi Richard, presented at ISFNT, to appear in Fus. Eng. Des. (2014)

Id 244

Abstract Final Nr. P4.112

A New Trial of TOKAMAK In-vessel Inspection Manipulator

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In this paper, we discuss the design and partial implementation of an in-vessel inspection manipulator in detail, which is considered to serve for China's Experimental Advanced Superconducting TOKAMAK (EAST). Besides the ordinary kinematic/dynamic constraints and specifications for a multiple degrees of freedom (DOF) manipulator suitable for EAST in-vessel inspection, this paper gives a comprehensive analysis on the necessity in design for the extreme environment, e.g., high temperature and high vacuum, and forms a practical design criterion accompanied by a comparison with relative manipulators and literatures. Based on our recent developed active cooling system, a specific proposal is explored, which employs ordinary commercial mechanical/electrical components only, as if the manipulator works in a normal temperature environment. This paper also emphasizes some challenging technical issues towards an implementation, such as, a special design of revolute joint structure, an optimization of thermal gradient/cooling path in the manipulator, a trade-off between large reachable workspace with large rotation angle of each joint and limited configuration for the cooling tubes going through joints, and so on. We develop an EtherCAT based real time control system connecting all the cameras, drivers and sensors, which achieves a robust closed-loop system and a clean cable aspect simultaneously. In the later part of the paper, basic mechanical tests on payload/accuracy and thermal experiments on partial joints are described. Evaluation on recent progress and future work towards a real test in the EAST is stated and expected.

Id 373

Abstract Final Nr. P4.113

Active Cooling System for TOKAMAK In-vessel Operation Manipulator

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In-vessel operation/inspection is an indispensable task for TOKAMAK experimental reactor. The internal information is then collected, such as the damage of first wall, 3D mapping, and localization. Also, specific tasks could be performed. A robot/manipulator is more capable in doing this than human being with more precise motion and less risk of damaging the ambient equipment. Although the inspection is carried out in the rest interval between plasma discharge experiments, an efficient accomplishment/diagnosis is expected. We demand the manipulator should be adaptable to rapid setup/response in high ambient temperature, and/or other extreme conditions, such as vacuum. In this paper, we propose an active cooling system embedded into such a manipulator as a solution. Cameras, motors, gearboxes, sensors, and other mechanical/electrical components in use could then be designed under ordinary conditions of temperature and so on. The cooling system cannot be a thermal shield only since the components are also heat sources in dynamics. We carry out a trial experiment to verify our proposal, then implement the whole active cooling system in detail based on theoretical analysis of heat transfer, discussion on varying design parameters, components and distribution. A closed-loop feedback control of temperature is carried out based on thermal sensors by adjusting liquid flux. High temperature endurance tests have been done on actively cooled manipulator joints superlatively. With the preliminary results, we believe that the proposal gives a way to robust and inexpensive design in extreme environment. Further work will concentrate on overall implementation and evaluation of this cooling system with the whole inspection manipulator together.

Id 427

Abstract Final Nr. P4.114

Trajectory planning of tokamak flexible in-vessel inspection robot

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Tokamak flexible in-vessel inspection robot is mainly designed to carry a camera for close observation of the first wall of the vacuum vessel, which is essential for the maintenance of the future tokamak reactor without breaking the working condition of the vacuum vessel.Based on EAST, a tokamak flexible in-vessel inspection robot is designded. In order to improve efficiency of the remote maintenance, it is necessary to design a corresponding trajectory planning algorithm to complete the automatic full coverage scanning of the complex tokamak cavity. With the forward, inverse kinematic and time-jerk optimal trajectory planning algorithms, two different trajectory planning methods , RS(Rough Scan)and FS(Fine Scan), according to different demands of the task, are used to ensure the full coverage of the first wall scanning. To quickly locate the damage position, the first trajectory planning method is targeted for quick and wide-ranging scan of the tokamak D-section part, and the scecond one is for careful observation. The process of robot scanning the vessel is simulated by OSG using the trajecoty planning data, The related curves of the joint angular displacement , angular velocity , angular acceleration and angular jerk are educed, and the curves are smooth and continuous. What’s more, both of the two different trajctory planning methods can ensure the full coverage of the first wall scanning with an optimal end posture. The method is tested on a simulation platform of EAST (Experimental Advanced Superconducting Tokamak) with the flexible in-vessel inspection robot , and the results show the effectivness of the proposed algorithm.

Id 607

Abstract Final Nr. P4.115

Vision-based online vibration estimation of the flexible in-vessel inspection robot

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The viewing system is installed at the end-effector of the inspection robot, which has a annular super-long structure, being used for damage detection of the first- wall in the inner chamber of tokamak.However, vibration may happend under the influence of external disturbance due to its structural features and material properties of elastic deformation. This affects the positioning accuracy, resulting in serious consequences unknow. Therefore, it is essential to implement vibration suppression. Usually, the vibration is measured by accelerometers, strain gauge, etc.But electronic components will not work properly in the nuclear environment. This paper proposed a vision-based method to estimate the vibration of end-effector of the robot. First of all we extract image feature points to get the information of high frequency vibration of manipulator, then vibration model is established on the bases of sine wave.Using the Short-Time Fourier Transform to estimate vibration parameters, and adding hanning window and selecting the effective length of window to increase accuracy and filter out noise. What's more, Phase difference correction is used to overcome inherent weaknesses of FFT. The algorithm is tested on the flexible manipulator designed for a Tokamak device in China, and experimental results validate the proposed strategy.

Id 607

Abstract Final Nr. P4.116

Vibration Control of a Flexible Manipulator Based on Visual Sensor for In-vessel Inspection

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A vibration control method is introduced for remote handling of a long-reach flexible manipulator into a fusion reactor. The in-vessel handling includes rest to rest operation for periodic inspection of the first wall inside the fusion reactor chamber. Because of the large operation space, the special shape of the chamber and energy saving, the manipulator is specially made long enough to cover the whole inspection area and lighter in weight to reduce motor power. But vibrations due to the flexibility of such configuration are one of the primary problems occurring over the remote handling, preventing from steady viewing and accurate positioning needed to examine, with the visual sensor equipped on the tip of the flexible manipulator, the extent of damage of bricks on the wall. Therefore, vibration suppression is essential issue to be solved urgently to ensure fine positioning and distinct images from the visual sensor. In addition, the fusion reactor with high operation temperature, ultra-high vacuum, significant residual magnetic field and radiation will constrain the usages of common sensors such as accelerometers or strain gauges. With visual inspection, a feedforward controller based on input shaping techniques combined with visual feedback control is proposed for suppression of residual vibration of the flexible manipulator as handled. Test of the target-specific controller is carried out on an experimental single-link flexible manipulator with a real time operating platform. As a consequence, the vibration behaviors are damped out during the operation of remote handling, thus superior viewing and positioning are reached. The performance of the vibration control scheme is evaluated in terms of level of vibration reduction and positioning. Both simulation and experimental results validate the effectiveness of the vibration control scheme.

Id 607

Abstract Final Nr. P4.117

Electromagnetic analyses and optimization for slit configuration of ITER blanket shield block

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Electromagnetic (EM) load is one of the most concerned design issues for the in-vessel components in the Tokamak device. For shield block (SB), quite a few of slits cut on the SB are designed to mitigate the EM load. In this study, a solid model of the standard blanket SB04 with vacuum vessel was developed, and FE analyses were performed to address the eddy current and EM force during major disruption (MD) with ANSYS code package. Key factors which potentially have great impacts on the eddy current were studied individually, such as the location, depth and number of the slits. In order to validate the analyses results, they are compared with currents on the passive structure from the output of DINA code. Current of plasmas which vary spatially and temporally are loaded as filament at each time point, and the magnetic surface of the field can be obtained to monitor the evolution of the plasmas configuration during MD, as Figure 1 shown. To understand the effects of cooling channels on the EM load, the case of SB without cooling channel are calculated to make comparisons. Two important conclusions can be made from this study. The first one is that cooling channels have a little impact on the EM load. This is because the cooling channels cut the eddy current only locally, and the global current loop is not affected. The second is that slit configuration is important to control the EM load. During MD, eddy current is induced on the horizontal and vertical plane, and it can be expected that radial slits have great impact on the EM load. However, the location, depth and number of the slits are also important.

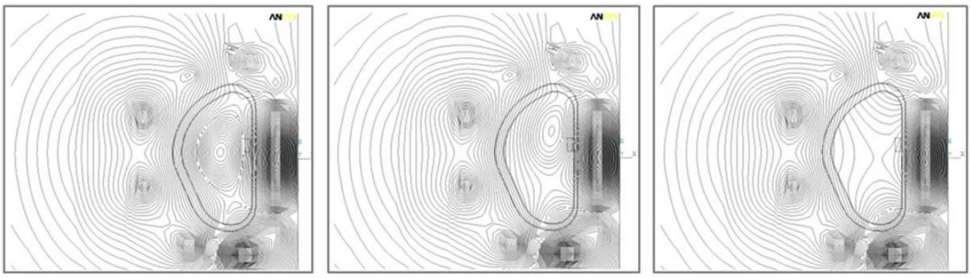

Figure 1 Plasmas configuration during MD: left, beginning; middle, t = 0.03s; right, end

Id 231

Abstract Final Nr. P4.118

Redesign of ITER divertor diagnostic rack: multiphysics simulations and experimental tests

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The attemts to comply the design of ITER divertor diagnostic rack with thermal, electromagnetic, strength and seismic requirements lead to the number of design iterations which were consequently considered and analyzed [1,2]. In the near to final design [3] the mirror fastening units were found to be the most critical construction elements, as they should provide the fixation if Si mirrors in the stainless steel frame. The difference in thermal expansion coefficients of these materials from one hand makes it to be not easy to achieve the reliable mirror fixation and from the other hand leads to the thermal stresses in fastening units and distortion of mirror optical surface which should be minimized. The performance of completely new design of the fastening unit is analyzed both numerically and experimentally. As a result pros and cons of the new fastening unit design are pointed in the paper. The full cycle of analyses for the final design of the diagnostic rack is also presented, including electromagnetic, thermal, stress, seismic and fatigue analyses. The results illustrate meeting of ITER requirements. [1] E.E. Mukhin, V.V. Semenov, A.G. Razdobarin, S.Yu. Tolstyakov, M.M. Kochergin, G.S. Kurskiev et al. The ITER divertor Thomson scattering system: engineering and advanced hardware solutions, Journal of Instrumentation, 7 (2012) C02063 [2] Victor S. Modestov, Alexander S. Nemov, Igor V. Buslakov, Aleksey I. Borovkov, Mikhail M. Kochergin, Eugene E. Mukhin et al. Engineering analyses of ITER divertor diagnostic rack design, Fusion Engineering and Design, 88 (2013), pp. 2038-2042

Id 1039

Abstract Final Nr. P4.119

Fusion reactor handling operations with cable-driven parallel robots

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Under development for several years now, cable-driven parallel robots (CDPR) are a new kind of robotic equipment which feature unique characteristics in terms of flexibility, stability and workspace. In its concept, it is basically a crane with inclined cables which allow control of all the degrees of freedom of the payload, and therefore stability of all the degrees of freedom, including rotations. Being based on cables reeled in and out by motors for its design, the workspace of a CDPR is only limited by the length of the cables, and the payload capacity related to the mass of the whole robot is very important. Besides, the control being based on kinematic models, the behavior of a CDPR is really that of a robot capable of automated trajectories or remote handling. Tecnalia is now preparing the industrial applications that are on the verge of being implemented in the real world for civil aircraft maintenance, civil works workshop assembly, workshop management and theatre performance, among others. Two prototypes have been built, the largest spanning 12m long, 10m wide and 6m high for a payload of 500kg, showing a repeatability of less than a centimeter in position and a degree in orientation. These prototypes have been used in order to demonstrate a number of user cases related to these applications. The present paper will give a presentation of a number of use case studies based on some of the assembly phases and remote handling actions as designed for the ITER tokamak. Based on the user cases already in place, the paper will show the opportunity of using CDPR for assembly of tokamak structural elements and coils, and for remote handling in a hot cell equipment from the tokamak and remote handling equipment such as force-feedback slave manipulators.

Id 134

Abstract Final Nr. P4.121

Analysis of the steady state thermal-hydraulic behaviour of the ITER blanket cooling system

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The blanket system is the ITER reactor component devoted to provide a physical boundary for plasma transients and contribute to thermal and nuclear shielding of vacuum vessel, magnets and external components. It is expected to be subjected to significant heat loads under nominal conditions and the design of its cooling system is particularly demanding. In fact, the mass flow rate distribution has to be optimized to ensure an adequate cooling to prevent any risk of critical heat flux occurrence while complying with pressure drop limits. At the University of Palermo a research campaign has been performed, in cooperation with ITER Organization (IO), to investigate the steady state thermal-hydraulic behaviour of the ITER blanket standard sector cooling system. A theoretical-computational approach based on the finite volume method has been followed and the RELAP5 system code has been adopted. Finite volume models of the most critical blanket cooling circuits have been set-up, realistically simulating the coolant flow domain as far as its geometrical, constitutive and hydraulic features are concerned. The steady state thermal-hydraulic behaviour of each cooling circuit has been investigated, determining its hydraulic characteristic function and assessing the spatial distribution of coolant mass flow rates, velocities and pressure drops under reference nominal conditions, to check that an adequate cooling is ensured avoiding an unduly high pumping power. Results obtained have indicated that most of the investigated cooling circuits should be able to provide an effective cooling to blanket modules under nominal steady state conditions, meeting IO requirements in term of pressure drop and velocity distribution. For specific modules where the results showed that the IO criteria or limits could not be met, potential variations to the cooling circuit lay-out have been proposed and their effects have been investigated to assess their possible influence on the optimization of the circuit thermal-hydraulic performance.

Id 186

Abstract Final Nr. P4.122

A computational procedure for the investigation of whipping effect on ITER High Energy Piping and its application to the ITER divertor primary heat transfer system

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The Tokamak Cooling Water System (TCWS) of nuclear facility has the function to remove heat from plasma facing components maintaining coolant temperatures, pressures and flow rates as required. Depending on thermal-hydraulic requirements, TCWS systems are defined as High Energy Piping (HEP) because they contain fluids, such as water or steam, at a pressure greater than or equal to 2.0 MPa and/or at a temperature greater than or equal to 100°C, or even gas at pressure above the atmospheric one. The French standards contemplates the need to consider the whipping effect on HEP design. This effect happens when, after a double ended guillotine break (DEGB), the reaction force, induced by the high velocity of the fluid and by the difference between inner and outer pressures, could create a displacement of the piping which might affect adjacent components. A research campaign has been performed, in cooperation by ITER Organization and University of Palermo, to outline the procedure to be adopted in order to check whether whipping effect might occur and assess its potential damage effects so to allow their mitigation. This procedure is based on the guidelines issued by U.S. Nuclear Regulatory Commission and namely ASME III, ANSI/ANS 58.2, Standard Review Plan (SRP), SRP Branch Technical Position adapted to TCWS piping load specification to limit the zones where leakages and DEGB (with associated whipping effect) could be postulated. The proposed procedure has been applied to the analysis of the whipping effect on the HEP of the divertor primary heat transfer system, using a theoretical-computational approach based on the finite element method. The verification of terminal ends and the application of loading conditions to determine intermediate rupture have shown that whipping effect may have a negative impact in the support performance and alternative solutions to their design have been proposed.

Id 349

Abstract Final Nr. P4.123

Numerical simulation of the transient thermal-hydraulic behaviour of the ITER blanket cooling system under the draining operational procedure

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Within the framework of the research and development activities supported by the ITER Organization (IO) on the blanket system issues, at the University of Palermo an intense research campaign has been performed, in close cooperation with IO, with the aim to investigate the thermal-hydraulic behaviour of the cooling system of a standard 20° sector of ITER blanket during the draining transient operational procedure. The research activity has been carried out following a theoretical-computational approach based on the finite volume method and adopting the RELAP5 system code. In a first phase, attention has been focussed on the development and validation of the finite volume models of the cooling circuits of the most critical and demanding modules belonging to the standard blanket sector, realistically simulating the coolant flow domain as far as its geometrical, constitutive and hydraulic features are concerned. Later on, attention has been put to the numerical simulation of the thermal-hydraulic transient behaviour of each cooling circuit during the draining operational procedure, to be carried out prior to each maintenance and or inspection operation for each blanket module by the injection of high pressure nitrogen. The draining procedure efficiency has been assessed in term of both transient duration and residual amount of coolant inside the circuit, observing that the former ranges typically between 40 and 120 s and the latter reaches at most 15 kg, in the case of twinned modules #6-7 cooling circuit. Potential variations to operational parameters (e.g. nitrogen injection pressure, nitrogen average pressure) and/or to circuit lay-out (inlet/outlet valves) have been proposed and investigated to optimize the circuit draining performances and to address important issues for the improvement of the Drying System design carried out at IO. In this paper, the finite volume models set-up are briefly described and the key results are summarized and critically discussed.

Id 348

Abstract Final Nr. P4.124

Concept design of divertor remote handling system for DEMO

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The work behind this paper takes place in EFDA remote maintenance system project (WPRM) for the Demonstration Fusion Power Reactor (DEMO), to follow ITER, confirming the capability of generating several hundred of MW of net electricity by 2050. The main objective of these activities is to develop an efficient remote handling (RH) system for replacing the divertor cassettes and the cooling pipes. In DEMO design one important consideration is the availability and short down time operations. This paper presents a concept design of the divertor RH system. The proposed divertor mover is a hydraulic telescopic boom driven from the transportation cask through the maintenance tunnel of the reactor. The boom is divided in three sections of four meters each, and it is driving and end-effector in order to perform the scheduled operations of maintenance inside the vacuum vessel. Taking advantage of the ITER RH background and experience, the proposed hydraulic RH system is compared with the rack and pinion system currently designed for ITER and object of simulations at Divertor Test Platform (DTP2) in VTT's Labs of Tampere, Finland. Pro and cons are put in evidence. Two alternative design of the End Effector to grip and manipulate the divertor cassette are also presented in this work. Both the concepts are hydraulically actuated, basing on ITER previous studies. The divertor cassette end-effector consists of a lifting arm linked to the divertor mover, a tilting plate, a cantilever arm and a hook-plate. Kinematic analyses have been studied in order to choose the best concept with less number of joints and mechanisms. The main objective of this paper is to illustrate the feasibility of DEMO divertor remote maintenance operations

Id 70

Abstract Final Nr. P4.125

Implementation of Simulation Lifecycle Management system using ITER Remote Handling case studies

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The work behind this paper takes place in the continuation of the EFDA's European Goal Oriented Training programme on Remote Handling (RH) "GOT-RH". The purpose of the work package 1.5 entitled: "Verification and Validation (V&V) of ITER RH system requirements using Digital Mock-Ups (DMU)" was to study and develop efficient approach of using DMU's in the V&V process of ITER RH systems. Complex engineering systems such as ITER facilities lead to substantial rise of cost while manufacturing the full-scale prototype. In the V&V process for ITER RH equipment, physical tests are a requirement to ensure the compliance of the system according to the required operation. Therefore it is essential to virtually verify the developed system before starting the prototype manufacturing phase. The paper describes the implementation of Simulation Life-cycle Management (SLM) system using ITER RH case studies for the system verification and validation. A proposed V&V process for ITER RH systems has been developed during the EFDA GOT-RH project. The developed process aims to enhance the use of DMU in the V&V process of ITER RH systems. In such process, the product and simulation data traceability is thereby essential all along the design process as well as the following phase such as operation, maintenance up to the decommissioning phase. Centralized around a SLM platform it will aim at decreasing the risk of developing further a non-conform system. Using such tool from the early ITER RH concept design phase will lead to reduce the risk of errors and thus increases the confidence of developing systems in accordance with the requirements all along the design process. The paper describes the advantage of using SLM platform in ITER design process and gives an overview of the limitations. Futures activities and expected results are also discussed.

Id 203

Abstract Final Nr. P4.126

Modelling and optimising method suitable for advanced development processes for the mechanical components located in the ITER vacuum vessel

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During our work we analysed and compared the limited general engineering requirements with the specific requirements of a fusion power plant. In this paper we aim to reveal methods that can decrease the time spent on the mechanical elements development increasing the efficiency of the process. The challenge in this scheme is the optimization of a large number and variety of components, as it is the case in typical mechanical engineering systems. Provides the opportunity of using the methods in different areas, so it may result a profitable industrial innovation, and is suitable for creating new ideas on the field of fusion engineering projects. The mechanical design considerations such as geometry compatibility and strength under dynamic and static loads were compared with the special requirements resulting from high risk factors such as electromagnetic loads, heating and radiation damage, radiation induced corrosion. Our goal is to create a method that survey complex demands, accomplish evaluation and the final integration of the inspected mechanical design. With this method we propose methodological recommendations for the mechanical design and the computer and physical simulation procedures.

Id 293

Abstract Final Nr. P4.127

50 Hz deuterium pellet injector for EAST tokamak

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50 Hz deuterium pellet injector for EAST tokamak Igor Vinyar1, Jiangang Li2, Alexander Lukin1, Jiansheng Hu2, Xinjia Yao2, Changzhen Li2, Yue Chen2, Pavel Reznichenko1 1PELIN, Saint-Petersburg, Russia 2Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China A new high frequency pellet injector has been developed for edge localized mode mitigation and plasma fuelling of the EAST tokamak. The pellet injector consists of two independent modules operating at 1-25 Hz each. Maximal injection frequency 50 Hz is achieved when two modules work in turn. Both modules are placed in one vacuum chamber and can be cooled with common or separate liquid helium supply line. Each module is equipped with a screw extruder for solid deuterium ice production in shape of a rod of 1.5x1.5 mm cross section at steady state mode for more than 1000 s. An innovative pellet fabrication and acceleration system has been designed and employed to reduce gas load on the extruders and to increase the pellet injection reliability. Pressurized helium gas is used for pellet acceleration to 150-250 m/s as well as for pellet fabrication from ice rod pushed away from the extruder. Pellets of 1.5 mm diameter and 1.2 mm, 1.5 mm, and 1.8 mm lengths can be fabricated using three types of the extruder nozzles which are replaceable on a day to day basis. Pellets are injected through two barrels which are inserted in one guide tube in a diagnostic chamber where its velocities are measured. Pellet-in-flight snapshots are also employed in the diagnostic chamber. Two guide tubes can be connected to the injector providing pellet injection into plasma from different directions (low or high field side). A prototype of 25 Hz pellet injection module has been already tested successfully. Long term injection of 1.5 mm size hydrogen pellets at 200-300 m/s with reliability over 93% was achieved.

Id 146

Abstract Final Nr. P4.128

Tritium control in fusion reactor materials

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In fusion reactors, tritium is bred by lithium isotopes present inside the blanket and extracted. However, tritium can contaminate the reactor structures, and can be lost by permeation into the environment. Tritium in reactor components should therefore be kept under close control throughout the fusion reactor lifetime, bearing in mind: 1) the risk of accidents; 2) the need for maintenance; 3) detritiation of dismantled reactor components before their re-use or disposal. A tritium permeation analyses code (FUS-TPC) was previously developed to perform the tritium transport analysis in fusion reactors, based on the mass balance equation for various chemical forms of tritium, coupled with a variety of tritium sources, sinks, and permeation models. An upgraded version of the code is presented here, being able to compute the tritium inventory in reactor components and materials at the moment of their dismantling, and to predict the results of detritiation processes. Concerning tritium transport and release during operation, the analysis showed that tritium losses and inventories in the reactor components are strongly dependent on the presence of tritium permeation barriers, on the performances of the Coolant Purification System and of the Tritium Extraction System. Concerning detritiation of spent materials, tritium surface decontamination techniques are modeled by the code. The residual T deeply trapped in crystalline imperfections is relatively immobile. Its removal before the material reprocessing is an unsolved problem. Electrolysis, iodine refining, arc and electron beam melting were considered and used for reprocessing and refinement of radioactive materials: the performance of each detritiation technique is predicted, according to material properties, tritium initial inventory and process conditions. The code is a useful tool in order to choose the most efficient techniques. A code validation with experimental results obtained in Russian and Italian laboratories is discussed, showing a good agreement.

Id 622

Abstract Final Nr. P4.129

Thermo-mechanical study of high heat flux component mock ups for fusion reactors

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Commercial infrared heaters have been proposed to be used in the HELOKA facility at KIT, to test a mock-up of the First Wall (FW), called Thermo-Cycle Mock-up (TCM), under stress loading comparable to those experienced by the Test Blanket Modules (TBMs) in ITER. Two related issues are studied in this paper, in the framework of the ongoing European project aimed at the design of the two EU TBMs: 1) the possibility to reproduce, by means of those infrared heaters, high heat flux conditions similar to those expected on the ITER TBMs 2) the preliminary thermo-mechanical analysis of the TCM, aimed at the definition of an experimental set-up (choice of experimental parameters and mechanical constraints) able to produce relevant stress conditions on the plate. For what concerns the heater experimental campaign, a model of the heaters has been developed and used to define an experimental matrix allowing the validation of the model. A comparison with experimental data shows that the model is suitable to perform the pre-test calculations for the TCM experimental campaigns. For what concerns the TCM plate, the thermal and structural analysis proves that the structure is able to withstand high thermal loads, even in the most constrained condition. In addition a possible testing strategy is supported by the analyses. Using highly constrained structures, but with reduced heat flux peak, it should be possible to reach stress levels for ratcheting and fatigue loads relevant to the ITER TBM, albeit in a modified set-up of heat flux and mechanical constraints.

Id 244

Abstract Final Nr. P4.130

Novel Technique for Synthesis of Nanocrystalline Li2TiO3 at Room Temperature by High Energy Ball Milling

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Lithium Titanate (Li2TiO3) is one of the most promising candidates among the tritium breeding materials because of its good tritium release capacity. We report first time the synthesis of nanocrystalline Li2TiO3 at room temperature without any external heat treatment or calcinations using a novel technique HEBM (High Energy Ball Milling). The X-ray analysis shows that the phase formation of single monoclinic crystal structure is formed room temperature. It was also found from TEM analysis that the synthesized powders were in nanocrystalline size 10nm and well match with X-ray data. The Complex Impedance Spectroscopy (CIS) studies show the presence of both bulk and grain boundary effects in the electrical properties of nanocrystalline Li2TiO3. The hopping frequency shifts toward higher frequency with increase in temperature.

Id 804

Abstract Final Nr. P4.131

Thermal analysis for K-DEMO breeding blanket and flow header using CFD and MARS code

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The thermal design of a breeding blanket for K-DEMO (Korean Fusion DEMOnstration reactor) is evaluated in previously study. The breeding blanket has a concept which is plate-type geometry consisting of stacked components in an arrangement parallel to the first wall. Heat sources are considered include surface heat load and nuclear heat generation inside the blanket. The pressurized water used as a coolant in the blanket is distributed and collected by flow headers located at the lower and upper parts of the breeding blanket. In the feasibility test, the thermal limits for a Li4SiO4 pebble bed, a beryllium pebble bed, and F82H are satisfied with the conceptual design respectively. In order to provide useful methodology for two-phase transient flow simulation, the MARS (Multidimensional Analysis of Reactor Safety) code is validated with CFD (Computational Fluid Dynamics) code for proposed breeding blanket design. The MARS model for the branch pipe which was not covered in previous study is validated. For that, pressure loss in flow header is identified using CFX-13 code. In first wall region, higher temperature was founded in CFD analysis compared to MARS code analysis in previous study. This is caused by treatment on temperature as bulk in MARS code. From this, one-side heating could yield an underestimation of temperature in the heating side and an overestimation of that in the opposite side. To solve this problem, the MARS modelling is further improved in order to minimize the error from one-side heating case.

Id 796

Abstract Final Nr. P4.132

Deuterium permeation in erbium oxide coating and RAFM steel via liquid Li-Pb alloy

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Tritium permeation through structural materials is one of critical issues in liquid lithium-lead (Li-Pb) blanket concepts in terms of an efficient fuel cycle and radiological safety. Fabrication of thin ceramic coatings on inner wall of metal ducts as a tritium permeation barrier is a promising solution. In recent years erbium oxide coatings have been investigated intensively; the hydrogen isotope permeation mechanism and high permeation reduction factors (up to 100000) have been achieved. However, these studies employed the gas-phase permeation system which might vary the permeation behavior through liquid Li-Pb alloy. In this paper, a liquid-phase permeation system has been constructed, and the deuterium permeation behavior in reduced activation ferritic/martensitic (RAFM) steels with and without the erbium oxide coating has been investigated. In addition, Li-Pb compatibility of the coatings has been simultaneously studied via a temporal change of permeability during the experiment. The erbium oxide coatings were prepared on RAFM JLF-1 disc substrates by filtered vacuum arc deposition with the deposition parameters same as the previous studies. The thickness of the substrate and the coating was 0.5 mm and 1.5 µm, respectively. The deuterium permeation apparatus was upgraded to measure deuterium permeation via also liquid breeders. A sample and two components were welded using electron beam to fabricate the sample assembly. Deuterium permeation measurements were performed with and without Li-Pb to compare permeation behaviors. Deuterium permeation experiments were performed for the JLF-1 substrate via gas and liquid phases. The permeability with liquid Li-Pb showed more than 50% higher than that without Li-Pb, indicating the removal of the natural oxide layer or the acceleration of deuterium permeation via Li-Pb. The deuterium permeation mechanism via Li-Pb will be clarified by the erbium oxide coatings because the coatings has been stable in Li-Pb for 500 h at 773 K.

Id 975

Abstract Final Nr. P4.133

Preliminary synthesize test of two-phase material for tritium breeder

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Demonstration power plant (DEMO) reactors require advanced tritium breeders with high lithium density and high stabilities under operating conditions. Lithium metatitanate (Li_2TiO_3) is one of the candidate materials among the proposed solid tritium breeders because of its good tritium release property and high chemical stability [1]. However, the tritium breeding ratios (TBRs) in Li_2TiO_3 pebbles is considered to be smaller than that of the other breeders due to its lower lithium density [2]. Therefore, lithium metatitanate with excess Li ($Li_{2+x}TiO_{3+y}$) has been recognized as a prominent candidate material for advanced tritium breeder because of its high Li density and high chemical stability [3]. On the other hand, a two-phase material have interesting characteristic, however, this material is not considered for tritium breeder yet. In this study, preliminary synthesize and characterization of the two-phase material of LI_2TiO_3 and Li_2SiO_3 (LTSO) was carried out. Although the Li vaporization characteristic of Li_2SiO_3 at high temperature is higher than that of $Li_{2+x}TiO_{3+y}$, the Li vaporization of the two-phase material LTSO is gradually decreased by the addition of LI_2TiO_3 (Fig.1). In Li/Li+Si ratio = 0.5, the amount of Li vaporization is very few. Theseresults showed that a two-phase material have new possibilities for advanced tritium breeder.

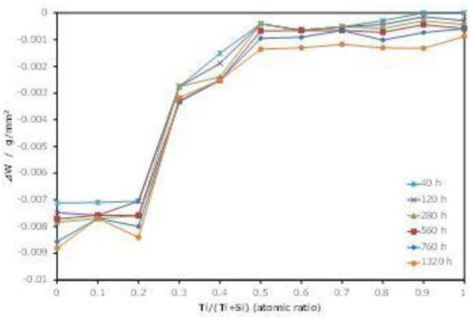

Figure 1 A weight loss from the sintered Li_2TiO_3-Li_2SiO_3 mixed material with varuous Li/Si ratio at 1173 K in 1%-H_2/99%-He.

[1] N. Roux, J. Avon, A. Floreancig, J. Mougin, B. Ravel, J. Nucl. Mater., 223?237 (2) (1996) 1431?1435. [2] M.Enoeda, Y. Kosaku, T. Hatano, T. Kuroda, N. Miki, T. Honma, M. Akiba, S. Konishi, H. Nakamura, Y. Kawamura, S. Sato, K. Furuya, Y. Asaoka, K. Okano, Nucl. Fusion 43 (2003) 1837?1844. [3] T. Hoshino, M. Yasumoto, K. Tsuchiya, K. Hayashi, H. Nishimura, A. Suzuki, T. Terai, Fusion Eng. Des., 82 (2007) 2269?2273.

Id 920

Abstract Final Nr. P4.134

Chemical compatibility of lithium meta-titanates with low-activation ferritic steel F82H

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Lithium meta-titanate with excess Li is proposed as a candidate tritium breeding material because of its high Li concentration. Its chemical reactivity is a great concern because the Li activity in the material is anticipated to be larger than that in the stoichiometeric lithium meta-titanate. In this study, the chemical reactivity of the material with low-activation ferritic steel F82H was investigated in comparison with that of the stoichiometric compound at 600 C and 800 C. Lithium meta-titanate pebbles with excess Li and stoichiometric composition (monoclinic-Li2TiO3 phase, 2.27 and 2.01 in Li/Ti ratio, respectively) were supplied by JAEA. Pebbles of each compound were heat-treated in contact with a F82H specimen supplied by JAEA under the contacting pressure of 3700 Pa in He+0.1%H2 at 600 C or 800 C for 100-800 h. The surface and the cross-section of each specimen after the treatment were analyzed by XRD, EPMA and SIMS. On the surfaces of lithium meta-titanates with both compositions, only small Li2FeO2 peaks were detected besides the peaks of monoclinic-Li2TiO3 phase by XRD, and no significant diffusion of the elements from F82H into the lithium meta-titanates was observed. On the surface of F82H specimen, on the other hand, LiCrO2 peaks were detected as well as F82H-derived peaks, and the cross-sectional observation of the F82H specimens by EPMA and SIMS gave some important information about the reaction and the diffusion of Li , O and Cr in the near-surface region. In case of stoichiometric compound, Cr diffused from the F82H specimen to form LiCrO2 on the surface. In case of excess Li compound, in contrast, Cr2O3 phase was formed in the deep region as well as LiCrO2 in the adjoining near-surface region. These results mean the diffusion of Li and O is enhanced presumably by their larger activities in the excess compound.

Id 410

Abstract Final Nr. P4.135

Three-dimensional flow measurement of a water flow in a sphere-packed pipe by digital holographic PTV

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A water cooled ceramic breeder for ITER and DEOM of a nuclear fusion reactor plays a significant role in the design of a blanket module. Pebbles of a ceramic tritium breeder are packed in a container of the blanket. Helium gas flows in a tritium recovery system. Investigation of the flow behavior is necessary in an actual environment of a facility where pressure drop takes place under a complex flow such as in case of the container for the pebble bed. For the development of a facility, it is necessary to be able to monitor fluid motion of a basic flow such as a sphere-packed pipe (SPP). In the present study, to discern the complex flow structures in SPP, digital holographic PTV visualization is carried out by a refractive index-matching method using a water employed as a working fluid. The water is chosen to be able to adjust its refractive index to match to that of the MEXFLON sphere with an index of 1.33. Hologram fringe images of particles behind the spheres can be observed, and the particles' positions can be reconstructed by a digital hologram. Consequently, 3-D velocity-fields around the spheres are obtained by the reconstructed particles' positions. The velocity between pebbles is found to be faster than that in other regions in the SPP.

Id 687

Abstract Final Nr. P4.136

Single pebble experiments to determine: applicability of Hertzian theory for pebble interactions, individual pebble modulus of elasticity, and pebble crushing in ensembles.

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DEM, as currently employed by members of the fusion community, begins with the assumption that each pebble is a perfectly elastic material that obeys Hertz's theory for normal interaction. This assumption impacts the magnitude of inter-particle forces predicted by the method. We scrutinize the Hertzian assumption with single-pebble crush experiments. We carefully record the force-displacement response and compare them to the non-linear forces predicted by a Hertzian pebble with bulk properties reported in literature. We found each pebble generally has a non-linear force response but with varying levels of 'stiffness' that qualitatively matched the curves from Hertz theory. Assuming Hertzian interaction, we then backed-out an elastic modulus for each pebble. We define a stiffness reduction factor, k, as the ratio of the pebble's elastic modulus to the sintered bulk value reported in literature. After determining the k value for every pebble in our batch, we discovered a Gaussian-like distribution that we attribute to the varying micro-structure of each pebble. The majority of the pebbles studied here had a concentration around k = 0.5; i.e. the pebble responded as if having a Young's modulus that was half the sintered bulk value. To incorporate the results into our DEM algorithms, we apply k values at random to pebbles with the single constraint that the final distribution satisfies the probability curves determined in experiments. Then, DEM simulations of pebble beds in oedometric compression are carried out to determine the contact forces, if they are above a threshold that would induce pebble crushing, and how many pebbles experience those forces. The extent of single-pebble crushing predicted in DEM is being compared against experiments on pebble beds with the same boundary conditions where we can measure the quantity of broken pebbles.

Id 757

Abstract Final Nr. P4.137

Water-LiPb interaction study

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The water-LiPb interaction implies a direct energy release, which leads to temperature and pressure increase, due to a combined thermal and chemical reaction and an indirect form of energy release, i.e. the hydrogen production, due to secondary chemical reactions involving the initial reaction products. Whether this reaction due to a water leak into the LiPb has the potential to breach the blanket box, or whether this risk can be ruled out is a question that impacts the concept design of the WCLL blanket. Review and understanding of the knowledge acquired in past studies, experimental works and numerical activities are needed in view of the renewed interest in the WCLL blanket concept and of planning of future R&D activities aimed at solving the safety issues connected with its design. The paper is divided into three parts. The first section of the paper addresses a review of studies carried out in the past, performed in US and EU to characterize the potential safety concerns associated with the use of PbLi eutectic alloy and water as breeding and coolant materials, respectively. Moreover, the BLAST and LIFUS5 experiments were presented to better understand the processes that caused the pressure peak and to investigate the consequences of LOCA accidents in liquid metals pools. The second part documents the post-test analysis by SIMMER-III code of BLAST test No. 5. The final section is aimed at presenting the main outcomes from the literature survey and from the experimental campaign. No code was found able to perform a satisfactory post-test analyses of separate effect experiments without engineering assumptions. Therefore, correlations that model the exothermic reaction and hydrogen production, and the availability of experimental data with more controlled initial and boundary conditions are needed for solving the WCLL blanket safety issues associated with the water-LiPb interaction.

Id 626

Abstract Final Nr. P4.138

Analysis of the thermo-mechanical behaviour of the DEMO Water-Cooled Lithium Lead breeding blanket module under normal operation steady state conditions

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Within the framework of DEMO R&D activities, a research cooperation has been launched between ENEA-Brasimone, the University of Palermo and the Commissariat à l'Energie Atomique to investigate the thermo-mechanical behaviour of the outboard equatorial module of the DEMO1 Water-Cooled Lithium Lead (WCLL) blanket under normal operation steady state scenario. The research campaign has been carried out following a theoretical-computational approach based on the Finite Element Method (FEM) and adopting a qualified commercial FEM code. In particular, two different 3D FEM models (Model 1 and Model 2), reproducing respectively the central and the lateral poloidal-radial slices of the WCLL blanket module, have been set up and optimized. A particular attention has been paid to the modelling of water flow domain, both within the segment box channels and the breeder zone tubes, to simulate realistically the coolant-box thermal coupling. A set of uncoupled steady state thermo-mechanical analyses have been carried out with the two models, supposing the module to undergo both 15.5 MPa coolant pressure on its cooling channels walls and thermal deformations due to the flat-top plasma operational state thermal field. Results obtained have shown that the EUROFER critical temperature of 550 °C is never overcome and that the maximum temperature of 509.7 °C is reached within the Model 1 Breeder Zone. Concerning the mechanical behaviour, the WCLL box experiences moderate Von Mises stress values, that remain under 200 MPa in Model 1 and below 350 MPa in almost the whole segment box of Model 2. Safety verifications, according to SDC-IC codes, are totally satisfied as far as Model 1 is concerned, while they are generally widely satisfied for Model 2, except for that relevant to the potential loss of ductility in the toroidal path located in the 2nd radial cell of poloidal-radial stiffening plates.

Id 187

Abstract Final Nr. P4.139

Blind scoping predictions for PAV experimental recovery efficiencies from a dynamic numeric model

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Blind numerical predictions and scoping calculations are common Quality Assurance logical procedure for synergistic R&D rationale results and predictive modelling tools developments. PAV [Permeator Against Vacuum] is a well-established conceptual proposal for the recovery of dissolved tritium bred inside lead-lithium (Pb15.7(2)6Li) liquid metal breeder. An experimental demonstration of PAV was launched by 2010 in the frame of Spanish Fusion Technology Programme TECNOFUS (see L. A. Sedano et al.; ISFNT-10 Portland, US) in collaboration with Industry /SENER. Its main goal was to qualify the H/D (then tritium) recovery efficiency of a PAV immersed in a closed lead-lithium few liters thermo-syphon loop. In the planned test hydrogen/deuterium is injected at know calibrated rates; building-up a concentration in the flowing liquid and permeating across PAV thin walls. Permeated hydrogen/deuterium is continuously measured by spectrometric gas leak-detector method. A direct measurement of the immersed PAV recovery efficiency is obtained and monitored on-line as the ratio between permeated and injected rates. A CFD/solutes blind scoping numerical calculation has been developed for the FUSKITE® experiment in order to anticipate parametrically the expected results as a further proof of predictive tools refinement quality. The numerical model represents the liquid metal buoyant flow along the loop and, specifically, inside PAV channels. Also, it provides a direct assessment of transient permeation flux and final steady-state recovery efficiencies depending on key sensitive parameters. As anticipated and explained the fitting of experimental transient release rate would confirm previously found Sievert´s constant values for tritium in lead-lithium eutectic.

Id 970

Abstract Final Nr. P4.140

Modelling the thermal-hydraulic dynamic behaviour of a generic fusion plant with dual coolant blanket and a supercritical CO2 power cycle using RELAP5-3D

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A generic fusion nuclear power plant has been simulated using RELAP5–3D. This code, developed by INL, has traditionally been used in the simulation of operational and accidental transients in fission nuclear plants. The aim of the work presented here is to demonstrate the suitability of the simulation capabilities of RELAP5–3D for fusion technology applications. The reference plant is inspired in the Spanish proposal for DEMO, designed under the Spanish Fusion Technology Program Consolider TECNOFUS. The hydraulic model of the plant includes the primary coolant systems in the breeding blanket, i.e. helium and LiPb in the Spanish dual coolant modular design (doble refrigerante modular, DRM); the divertor's cooling system; the conversion system (supercritical recompression CO2 power cycle); the ultimate heat sink (water from nearby river or lake); and an intermediate thermal energy storage system to allow continuous electrical energy export to the grid. Heat fluxes among the different fluids and circuits, along with the heating supplied by the plasma are simulated by means of RELAP5 heat structures. The CO2 power cycle includes compressors, turbine and heat exchangers (Printed Circuit type). The model has been equipped with basic control features, whose parameters have been adjusted by means of preliminary calculations. A few transient calculations have been run, that have been useful to understand de dynamic behaviour of the plant and to demonstrate the capabilities of code and model. The use of system codes to simulate fusion reactors can be helpful in several ways: models can predict the coupled behaviour of the several subsystems, they can be used to adjust the parameters of the control systems needed for the operation; models may help understanding real plant transients (their predictions reach locations where no sensors exist); finally, thermal-hydraulic models allow fast sensitivity analysis, thus complementing the analysis done with other tools.

Id 1005

Abstract Final Nr. P4.141

Formation and annihilation of radiation defects and radiolysis products in modified lithium orthosilicate pebbles with addition of titania

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Lithium orthosilicate pebbles with 2.5 wt% of silica are the European Union's designated reference tritium breeding ceramics for the Helium Cooled Pebble Bed (HCPB) test blanket module (TBM). The reference pebbles feature appropriate tritium breeder parameters, i.e. high lithium density, high melting point and good tritium release behaviour. However, latest irradiation experiments showed that pebbles may crack and form fragments under operation conditions as expected in an HCPB TBM. Therefore it may be favourable to change the ceramic composition and to replace the excess of silica by titania to obtain lithium metatitanate as a second phase. The aim of this research was to investigate the formation and annihilation of radiation defects (RD) and the radiolysis products (RP) in the modified lithium orthosilicate pebbles with additions of titania. Modified lithium orthosilicate pebbles with different amounts of titania were selected for investigation together with reference pebbles. Irradiation was performed with accelerated electrons up to 5 GGy absorbed dose at 380-670 K in dry argon. The formed and accumulated paramagnetic RD and RP were analyzed with the electron spin resonance (ESR) method. To investigate the thermal stability and the annihilation of the localized RD and RP, the irradiated pebbles were thermally treated up to 770 K for 20 min in vacuum furnace. Using ESR spectroscopy it has been determined that in the modified lithium orthosilicate pebbles several species of paramagnetic RD and RP are formed. The obtained results indicate that by replacing the excess of silica with equal amounts of titania, the total concentration of paramagnetic RD and RP in the modified pebbles slightly decreases. The accumulated RD and RP in the modified pebbles practically annihilates after thermal treatment up to 770 K in vacuum. After 770 K in the ESR spectra only one, quite narrow signal ($g=2.001\pm0.003$, $\Delta H=0.4\text{-}0.5$ mT) was detected.

Id 150

Abstract Final Nr. P4.142

Spatially and temporally dependant transmutation in HCPB breeder blankets

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High energy neutrons produced through fusion reactions in a future magnetic confinement fusion reactor will interact with the structural and tritium breeder material in the blanket. These interactions result in significant transmutation reactions within the different blanket materials including the important tritium breeding reactions. The reaction rate for a given reaction depends on the neutron spectrum, neutron flux and material composition. The changing composition within the material will result in a time dependant reaction rate. The neutron flux and spectrum varies as a function of position both around a tokamak and depth through a blanket module. The repercussions of a spatially and time dependant reaction rate on the tritium production of a HCPB breeder blanket is discussed in this paper. A DEMO like tokamak with 50 homogeneous breeder blanket modules comprising of reduced activation steel, helium gas coolant, beryllium and lithium orthosilicate with enriched lithium 6 content was modelled in MCNP6. Each blanket module was segmented into 40 radial sections so that time dependant reaction rate as a function of depth could be analysed. In order to perform time dependant simulations the fusion specific depletion code FATI was used which couples MCNP6 with FISPACTII. The simulated results show how the tritium production as a function of depth through the blanket varies over its expected lifetime. The overall tritium production of the blanket decreases as the Li6 and Li7 are burnt up. The breeder zones at the inner surface of the breeder blanket show particularly large reduction in tritium production over time. Lithium 6 burnup within the inner most surfaces of the breeder blanket is considered the main contributing factor to the decreasing tritium production of the whole breeder blanket.

Id 48

Abstract Final Nr. P4.143

Thermal-hydraulic Analysis on Water-cooled Breeder Blanket for CFETR

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China Fusion Engineering Test Reactor (CFETR) is an ITER-like superconducting Tokamak reactor. Its major radius is 5.7 m, minor radius is 1.6 m and elongation ratio is 1.8. Its mission is to achieve 50–200 MW of fusion power, 30–50% of duty time factor, and tritium breeding ratio not less than 1.2 to ensure the self-suffciency. As one of the breeding blanket candidates for CFETR, a water cooled breeder blanket with superheated steam is proposed and its conceptual design is being carried out. In this design, sub-cooling water at 260ºC under the pressure of 7MPa is fed into first wall (FW). Then the water is fed into cooling plates to superheat up to about 500ºC. In this paper, a one-dimensional analytical program was used to calculated relevant thermal-hydraulic parameters. Under the condition of 200MW fusion power of CFETR, thermal-hydraulic parameters such as flow velocity, onset of boiling, fluid temperature, heat transfer coefficient, wall temperature and pressure drop were analyzed as well as the temperature distribution of breeding blanket.

Id 331

Abstract Final Nr. P4.144

Thermal analysis of the pebble beds for CFETR helium cooled solid breeder blanket

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Pebble beds are the key components in the solid tritium breeder blanket and subject to severe conditions, such as high temperature and irradiation. As one of the candidates for CFETR tritium breeding blanket, a kind of helium cooled solid tritium breeder blanket was proposed. The blanket uses the pebble beds of lithium ceramics (Li4SiO4 or Li2TiO3) and beryllium as tritium breeder and neutron multiplier, respectively. The thermal stability of the pebble beds directly affects the performance of tritium breeding and the safety of the pebble beds. Therefore, the thermal analysis of the pebble beds is of vital importance for a reliable design of the helium cooled solid breeder blanket. The thermal condition of the pebble beds are mainly impacted by two factors, the coolant thermal hydraulic condition and the pebble structure including material properties, pebble size, packing factor, and even purge gas. In this paper, the effects of different geometries and roughness of the cooling channel on the thermal hydraulic performance were assessed. The results indicated that the circular cooling pipes can remove the heat more effectively due to its larger contact area with the pebble beds. Increasing the roughness of the cooling channel a little improves the heat transfer while having little imparts on the pressure drop. The effects of the material properties, pebble size, packing factor were evaluated. Based on the investigation, an optimised scheme of cooling channel and packing pebble bed were recommended.

Id 173

Abstract Final Nr. P4.145

Design and research on the measurement platform of the effective thermal conductivity for Li4SiO4 and Li2TiO3 pebble bed

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China is carrying out the conceptual design of Chinese Fusion Engineering Testing Reactor (CFETR), and the Helium Cooled Pebble Bed (HCPB) Blanket concept is one of the main choices for tritium production.Li4SiO4and Li2TiO3 are the reference breeder materials for the HCPB Blanket concept. In the HCPB blanket breeding pebbles with the diameter range of 0.6–1.2mm are placed between two cooling plates. Accordingly, effective thermal conductivity of pebble beds will need to be determined for the heat transfer calculation. Measurements of the heat transfer parameters of Li4SiO4 and Li2TiO3 pebble beds are being performed at the University of Science and Technology of China (USTC).Two measurement methods are being used. One is the steady state method with the use of thermalcouples to measure the temperature distribution of the pebble bed.Another is the use of transient thermal probe method using the temperature variation of the thermal probe and Monte Carlo inversion method to calculate the heat transfer parameters of the pebble bed. This paper will report on the progress of these measurements at different purge-gas flow pressures,velocities, and the use of different purge gases.

Id 482

Abstract Final Nr. P4.146

Shutdown dose rate analyses for the WCCB CFETR

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The CFETR is Chinese next generation magnetic fusion energy (MFE) device based on ITER physics and technology but some optimized performances such as plasma burning time and TBR, aimed to maintain long and steady-state operation. The water-cooled ceramic breeder blanket (WCCB) developed by ASIPP is one of the candidate blankets for CFETR and the initial CAD mode has been achieved. The performance of CFETR equipped with WCCB will be assessed by the necessary neutronics analyses. The shutdown dose rate calculation then is ongoing in terms of the framework of CFETR utilizing a rigorous-2-step (R2S) approach. The R2S approach first needs the neutron flux with suitable energy groups based on Monte Carlo method. The nuclide inventory next will be carried out by the European activation code FISPACT with the nuclear cross section data FENDL-2.1. The obtained photon spectrum by the foregoing inventory calculation then will be implemented in the subsequent photon transport calculation to get the dose rate after shutdown. It's worth mentioning that the input file employed by neutronics Monte Carlo calculations will be generated form the CAD data by the McCad conversion software which is developed by KIT/INR. The initial result of the dose rate 12 days after shutdown in the blanket port extension shows a higher value than the designed limitation of 100 μSv/h. This may has a relationship with the designing of WCCB and later the optimized programs will be proposed step by step.

Id 574

Abstract Final Nr. P4.147

Optimization of coolant flow distribution for CFETR helium cooled solid breeder blanket

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Evaluation of temperature fields on the blanket structures is an indispensable step in the design process because the application of structural materials are confined by the narrow temperature windows. To obtain a reliable blanket scheme, accurate flow control in the blanket should be assured. As one of candidates for Chinese Fusion Engineering Test Reactor (CFETR) tritium breeding blanket, a kind of helium cooled solid tritium breeder blanket was proposed. The blanket is a steel box formed by several plates with internal cooling channels. The coolants in the channels are distributed from the manifolds in the back of the blanket. In this contribution, the flow characteristics of the helium gas in the manifolds of the solid blanket were simulated using computational fluid dynamics code. The flow rates of each cooling channel were obtained and it was found that the coolant cannot be distributed uniformly into the steel plates for extracting heat due to the various channel shapes or dimensions, the different locations of the inlets as well as the obstruction of the structures in the manifolds. As a result, the temperatures in some locations of the blanket components were beyond the material temperature limits. To optimize the flow distribution, several rib-like structures were adopted to guide the flow and the effectiveness of the method were verified.

Id 703

Abstract Final Nr. P4.148

Preliminary Neutronics Design and Analysis of Helium Cooled Solid Breeder Blanket for CFETR

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Chinese Fusion Engineering Test Reactor (CFETR) is a test tokamak reactor being designed in China to bridge the gap between ITER and future fusion power plant. Tritium self-sufficiency is one of the most important issues for CFETR and the tritium breeding ratio (TBR) is designed for at least 1.2. As one of the candidates, a helium cooled solid breeder blanket for CFETR superconducting tokamak option was proposed. In the concept, radial arranged U-shaped breeding zones are adopted for higher TBR and more simple structure. Ceramic pebbles Li4SiO4 and beryllium are used as breeder and neutron multiplier respectively, and the blanket structures are made of RAFM steel. In this work, three-dimensional neutronics design and analysis of the blanket were performed with the Monte Carlo neutron-photon transport code MCNP and IAEA data library FENDL-2.1. A 11.25° symmetric torus global reactor model was carried out for calculations. The 6Li enrichment, breeder ceramic and beryllium pebble beds configuration were optimized in both inboard and outboard blanket for improving the TBR. Meanwhile, the shielding effectiveness were assessed and the nuclear heating in the blanket was obtained as the input data for thermal and structure analysis. The results showed that the blanket can well meet the tritium self-sufficiency target and the neutron shield can satisfy the design requirements.

Id 575

Abstract Final Nr. P4.149

Thermal and structural analysis of the first wall for CFETR helium cooled solid breeder blanket

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Due to the tough conditions of high heat flux from plasma as well as intense irradiation from high-energy neutrons, the structure design of the first wall (FW) directly affect the safe operation of the whole blanket and performance evaluation of the FW is of great importance. As one of the blanket candidates for Chinese Fusion Engineering Test Reactor (CFETR), a conceptual design of helium cooled solid breeder blanket was proposed. In the blanket, a trapezoidal U-shaped FW was adopted for decreasing the gaps between the blankets. In this paper, to verify the effectiveness of the design, a fluid-solid coupled code was used to analyze the temperature and stress distribution in the FW under normal load and ultimate load. Furthermore, the influences of different FW geometrical configuration, including the curvature of U-shaped structure, the chamfer size and roughness of square coolant channel, on the temperature field and thermal stress distribution were investigated. Based on the results obtained, an optimized scheme was suggested, which can effectively reduce the maximum temperature and stress of FW structural material.

Id 704

Abstract Final Nr. P4.150

Operating modes of electrochemical H-concentration probes for tritium sensors

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Solid state electrolytes have been used as sensors components for over 40 years. These materials have several advantages in the handling/processing of molten metals: the conductivities of solid electrolytes increase with increasing temperature, the output of solid electrolyte based sensors is determined by the thermodynamic properties of the molten metal / reference electrode and finally, solid electrolytes are generally stable compounds that can withstand the harsh chemical environment in molten metals. A number of studies on proton conducting solid state electrolytes have been carried out due to considerable interest for applications in hydrogen gas sensors, hydrogen pumps, solid oxide fuel cells, etc. Potentiometric hydrogen sensors using different solid-state electrolytes have been designed and tested at the Electrochemical Methods Lab at Institut Quimic de Sarria (IQS). The most promising element (Sr(Ce0,9-Zr0,1)0,95Yb0,05O3-?) has been selected for this work in order to evaluate the sensor performance at different hydrogen concentrations in two different operating modes: amperometric and potentiometric. In addition, the sensor response has been evaluated at different working temperatures (450, 575 and 675°C). Both operating modes showed a very short response time when the hydrogen concentration was changed in the environment (10 minutes). These experiments proved that the sensor is able to follow fast changes in the hydrogen concentration of the system. Finally, it worth to mention that when the sensor was used in a potentiometric mode better results were obtained at low hydrogen concentration levels. However, when the sensor was used in an amperometric mode better results were obtained at a high hydrogen levels.

Id 1060

Abstract Final Nr. P4.151

Failure of a Lithium-filled Target and Some Implications for Fusion Components

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An experimental system for testing a lithium-helium heat exchanger at Sandia National Laboratories included the heat exchanger, a preheater for helium and another for lithium. The preheaters were to be heated separately by two electron beams to provide independent control of the temperatures of the helium and lithium streams going into heat exchanger. A lithium loop supplied molten lithium for the experiment. During the final preparation for the experiment, and before the helium system was pressurized or use of the electron beams began, unexpected and rapid failure of the lithium preheater occurred when lithium initially flowed into the preheater. This paper first describes the analysis of the failed lithium preheater and then some implications for fusion systems. Neutron radiography showed a network of long cracks. Metallurgical examinations showed features that indicated liquid metal embrittlement. Use of lithium in breeding materials is implicit for D/T fusion, and proposed future applications include lithium-cooled stainless steel first walls (e.g., in LIFE) and liquid breeding blankets with lithium-lead in ferritic alloys. And, liquid lithium surfaces are already of interest for plasma facing components in the near term applications in FTU, NSTX and EAST. However the literature available to guide fusion researchers regarding applications with lithium seems wanting in regard to both (a) choices of materials for fusion applications and (b) conduct of operations in establishing laboratory experiments as we develop experiments to investigate blankets with integrated first walls and plasma facing components with surfaces of liquid lithium.

Id 997

Abstract Final Nr. P4.152

Why is 9%Cr an optimal concentration for RAFM steels? A physical explanation.

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Reduced-activation ferritic-martensitic (RAFM) steels, such as Eurofer and F82H, are the main candidate structural materials for fusion reactors. They are 9%Cr-1%W steels that offer a number of desirable properties for use in a fusion environment, namely good thermal behaviour and good radiation-resistance, especially low swelling. The choice of ~9%Cr was dictated among other considerations by the requirement of having the lowest possible radiation-induced embrittlement at low operating temperature (~300°C), which would correspond to the water-cooled DEMO design. In general, keeping radiation embrittlement and swelling under control are both important choices, for any reactor design. However, so far the choice of 9%Cr was not supported by any understanding of why, physically, this Cr content provides the lowest radiation embrittlement possible, while guaranteeing low swelling. Recent studies that combined in-depth microstructural characterisation of Fe-Cr alloys of different Cr content with the development of advanced atomistic and dislocation models allowed the physical mechanisms responsible for these effects to be identified. The origin of these effects is the reduction of the mobility of dislocation loops due to the Cr content. This reduction leads on the one hand to enhanced recombination between vacancies and interstitials, thereby reducing swelling as compared to alloys that do not contain Cr, and leads also to a decrease of the density of visible loops with increasing Cr content, thereby reducing the corresponding radiation hardening and subsequent embrittlement. However, above 9%Cr the precipitation of α' reverses the trend, leading to higher radiation hardening and embrittlement. Moreover, the decoration by Cr atoms and other solute atoms of dislocation loops, both visible and invisible to the electron microscope, is in fact the main cause of radiation hardening and subsequent embrittlement in high-Cr steels. In this paper, we shall present the experimental results and models that quantitatively support these conclusions.

Id 364

Abstract Final Nr. P4.153

D desorption behavior for Fe2+ damaged W with various damage concentration

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The deuterium retention and desorption behavior is thought to be one of key issues for feasibility studies of plasma facing materials. Recently, tungsten (W) is considered as a promised material. In our previous studies, the D retention for 0.025 dpa neutron irradiated W has clearly different from that for un-irradiated W and large desorption stage was found at higher temperature above 800 K, leading to the retention characteristics would be controlled by the damage introduced during the plasma operation in fusion reactor. However, the detail mechanism of the formation of stable trapping sites in W is not well-understood. In this study, the damaged W was formed by Fe2+ implantation and D retention behavior was studied. The simulation of TDS spectra was also applied to explain the activation energies of D trapping sites in damaged W. The damage was introduced for W by 6 MeV Fe2+ implantation with the damage concentration of 0.0003 – 1.0 dpa at room temperature. Thereafter the 1 keV D2+ was implanted with the fluence of 5.0×1021 D+ m-2. The D2 TDS spectra showed that the D desorption stages consisted of three stages at around 400 K, 600 K and 800 K, corresponding to the desorption of D adsorbed on the surface or tapped by dislocation loops, that trapped by vacancies and that by vacancy clusters (voids), respectively. The retention of D trapped by vacancies and voids were increased as increasing the damage concentration. In especially, the D retention for voids was largely increased above 0.03 dpa. In addition, the desorption temperature was shifted toward higher temperature side indicating that D trapping / detrapping by various trapping site influences on the D diffusion toward the surface. These desorption behavior was simulated by the diffusion and trapping / detrapping model. It can be said that the density of trapping sites would control the D desorption temperature and shape of TDS spectra.

Id 279

Abstract Final Nr. P4.154

Spark plasma sintered optical ceramics for fusion applications

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Development of optically transparent ceramics can be a solution for neutron resistant diagnostic systems of future fusion power plants. Three well approved selection criteria can be used for finding radiation resistant compounds – (i) high ionicity of bonding, (ii) complex large unit cell and (iii) stoichiometric lattice must contain some empty positions, so-called stoichiometric vacancies. Also, it was shown earlier in several works that recombination of Frenkel pairs of defects on grain boundaries can provide extreme radiation resistance of ceramics. Many of alumina-based compounds fulfil most of these criteria and attracted considerable attention as candidate materials for fusion applications. In this study, high density ceramics starting from high-purity ultra-porous alumina (UPA) were prepared using Spark Plasma Sintering (SPS) technique, which is acknowledged method for obtaining dense, fine grained and optically transparent ceramics. UPA monoliths were obtained by oxidation of high-purity aluminium plates through a liquid mercury-silver layer in a controlled atmosphere [1]. Pure and trimethylethoxysilane treated UPA monoliths were sintered in air to form γ-alumina polymorphs which after grinding were used in SPS processing. The influence of SPS processing parameters on the phase composition, microstructure, density and transparency of the alumina-based ceramics were analysed. Transparency at room temperature in the spectral range of 200-5000 nm was tested. Electron beam (15 and 150 keV) irradiation has been used to get preliminary data on radiation resistance of synthesized ceramics. Low luminescence yield is required for the diagnostic windows, therefore cathodoluminescence in the spectral range of 150-1600 nm was investigated at various temperatures in order to reveal nature of luminescence centres. Present study reveals that starting from gamma-alumina dwell temperatures higher than 1350 0C are needed for obtaining a high density ceramic with grain size < 1.5 μm. [1] J.L. Vignes, et al, J. Mater. Sci. 43 (2008) 1234-1240.

Id 502

Abstract Final Nr. P4.155

Novel approach for joining Ceramic Matrix Composites for very high temperature applications

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Advanced Ceramic Matrix Composites (CMC), such as SiC/SiC, C/SiC and C/C, are promising candidates for components and subsystems in different applications at extreme working conditions. Potential applications of CMC are in propulsion, defense, aerospace and also in fusion and advanced fission energy applications due to their excellent radiation resistance A critical issue for a wider use of CMCs is the development of cheap, easily applied and reliable joining methods to assemble large components into more complex structures. e.g ultra-high temperature ceramics can be combined with CMCs for the fabrication of systems with enhanced performance under the most severe conditions. Thus, advances in joining science and technology for both CMC-metal and CMC-ultra high temperature and oxidation resistant ceramics are important in order to exploit the benefits of these advanced materials in a wide range of applications. The work presented in this conference refers to a novel approach for joining of CMC for ultra high temperature applications. The joining is produced by a filler metal based on a MAX-Phase (Ti3SiC2) fabricated by self-propagating high temperature synthesis. The process parameters of the joining method, the microstructure obtained and the performance under thermomechanical tests are discussed. The good results obtained under thermal shocks of the structure combined with the good thermal, mechanical, and crystal stability of the joined materials points to potential applications in the field of nuclear fusion

Id 317

Abstract Final Nr. P4.156

Impact properties of dissimilar weld joint between F82H low-activation ferritic steel and type 316L stainless steel after neutron irradiation

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Reduced activation ferritic/martensitic (RAFM) steels and type 316L stainless steels (SS316L) are the structural materials for fusion blanket and out-vessel components, respectively. Development of the joining techniques for RAFM steels to SS316L is essential to connect the fusion blanket to the out-vessel components. In the present work, a dissimilar joint between a RAFM steel, F82H-IEA, and SUS316L steel was fabricated. Impact properties of the joint were evaluated to discuss the resistance to neutron irradiation and deformation mechanisms under irradiation condition. The joint of F82H and SS316L was fabricated by using electron beam welding (EBW). Miniature Charpy V-notch specimens with a size of 1.5 x 1.5 x 20 mm, a notch depth of 0.3 mm and a notch root radius of 0.08 mm, were machined from the joint. The notch was placed at the base metal of F82H (F82H-BM), the base metal of SS316L (SS316L-BM), the center of the weld metal (WM) and inside the heat-affected zone on the F82H side (F82H-HAZ). Since the WM and the HAZ were much harder than the BMs, post-weld heat treatment (PWHT) was conducted at 953 K for 1 hr to recover the hardness to the level of the BMs. Neutron irradiation was carried out at 573 K with a dose of 0.1 dpa in Belgian Reactor II (BR-II). Hardness of F82H-BM, F82H-HAZ, SS316L-BM and WM before irradiation was 228 HV, 257HV, 207 HV, and 220 HV, respectively, whereas 230 HV, 300HV, 230 HV and 270HV after the irradiation. It is noted that limited area of F82H-HAZ with a size of about 50 micron showed a very large hardness of 400 HV after irradiation. Both WM and even F82H-HAZ exhibited better impact energy after the irradiation compared with F82H-BM. The irradiation hardening above mentioned did not degrade the impact properties of the joint. The mechanisms for the irradiation hardening and deformation behavior of the joint are discussed in the present study.

Id 468

Abstract Final Nr. P4.157

Corrosion of steels in molten tin (Sn) and tin lithium alloy (Sn-Li)

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Tin lithium alloy (Sn-20Li) has several attractive thermophysical properties of high thermal conductivity, heat capacity and low vapor pressure, when compared with those for Pb-Li alloy, though the tritium breeding performance of Sn-20Li alloy is lower than that of Pb-Li alloy. Liquid Sn and Sn-20Li alloy can be coolant of liquid divertor due to their favorable properties. The compatibility data of Sn and Sn-Li alloy is limited to exclude Sn-20Li alloy from the candidates of liquid breeders. The purpose of the present study is to investigate the compatibility of steels in liquid Sn and Sn-Li alloy. The compatibility of reduced activation ferritic martensitic steel JLF-1 (JOYO-HEAT) and 316 type austenite steel (SUS316) was investigated by means of static corrosion tests. The specimens of the JLF-1 steel (Fe-9Cr-2W-0.1C) and SUS316 (Fe-18Cr-12Ni-2Mo) were immersed in liquid Sn and Sn-20Li alloy under Ar atmosphere at 873K for 250 hours and 750 hours. The corrosion characteristics of the steels in liquid lead (Pb) and gallium (Ga) were also investigated for comparison with those in Sn and Sn-20Li. The weight loss data of the specimens and the results of the metallurgical analysis by EPMA were obtained in the present study. These results indicated that the corrosion of the JLF-1 and the SUS316 steels in liquid Sn and Sn-Li alloy was commonly caused by the alloying reaction between Sn and the chemical components of the steels. The thickness of the alloying layer of the JLF-1 specimen in 250-hr test was approximately 100µm, and the weight loss of the specimen was 0.978x103 g/m3. The results of the present study indicated that the corrosion of the steels in Sn and Sn-Li was caused by the alloying reaction, and the corrosion was intensive rather than that in Pb and Pb-17Li.

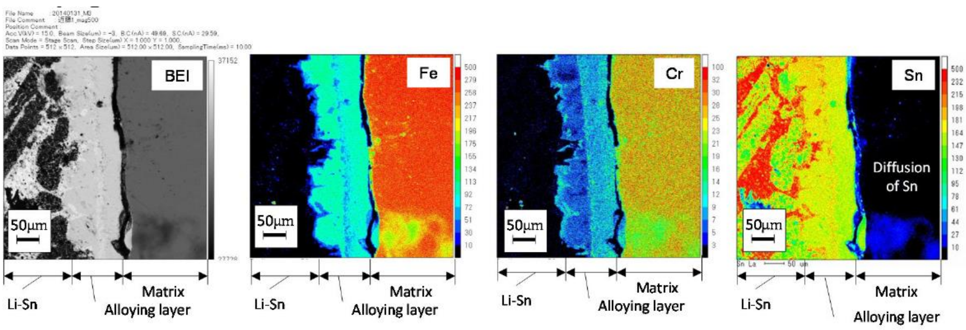

Fig. 1 Results of EPMA analysis of surface cross section in JLF-1 specimen exposued to Sn-20Li for 250 hours

Id 853

Abstract Final Nr. P4.158

Study on fabrication method of lithium alloy with metal grains

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Fabrication method of lead-lithium alloy (Pb-Li) and tin-lithium alloy (Sn-Li) using small metal grains of Li, Pb and Sn was investigated. The purpose of the present study is to investigate the fabrication methods of the Li alloys at low temperature around the melting point. The dissolution of non-metal impurities into the alloys can be supressed by this method due to the lower chemical activity of Li and lower solubility of the non-metal impuries in the metals at lower temperature. The Pb-Li and Sn-Li alloys with various Li content were fabricated with the metal grains of Li, Pb and Sn. The diameter of the metal grains was approximately 2.5mm. The quantity of the fabricated alloy was approximately 30cc. The fabrication was performed under Ar atomosphere at approximately 623K. The concentration of Li in the fabricated alloys were determined by the measurement of their melting point and the chemical analysis with inductivity cunpled plasma atomic emission spectrometry (ICP-AES). The alloys were fabricated by a melting and mixing with three types of initial mixing states (a, b and c), where the Li metals are placed as a lminer state between the Pb or Sn grains or after mixing well with Pb or Sn grains (Fig.1). In the mixing state (a), the Li grains were simply placed over the Pb or Sn grains. In the mixing state (b), the Li grains were placed in the middle layer of the Pb or Sn grains. In the mixing state (c), the Li grains were randomly mixed into that of Pb or Sn grains. In the experiment started with the mixing states (a) and (b), the fabrication of Li alloy was a failuer due to the segregation of Li rich alloy. In the states (c), the Li alloys were fabricated though the Li concentration in the alloy was lower than that of the target composition. Small fire was generated from the grains during the melting and the mixing (Fig.2). The low Li concentration was due to the oxidation reaction of the Li metals with the oxides on the Pb or Sn grains. The Li concentration in the alloy was adjusted to the target composition by a melting and a mixture of two kinds of Li alloys having different Li concentration.

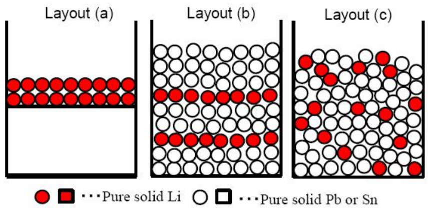

Fig. 1 Initial mixing state of metal grains

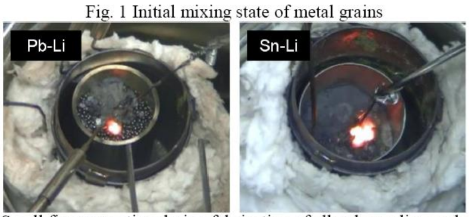

Fig. 2 Small fire generation during fabrication of alloy by meling and mixing

Id 734

Abstract Final Nr. P4.159

Surface grain orientation and 2D texturing of tungsten by laser triggered local melting and rapid solidification

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Driven by the needs of plasma-facing materials in the divertor region in the International Thermonuclear Experimental Reactor (ITER) project, tungsten with full density and high thermal conductivity is highly desired, which is beneficial for avoiding thermal shock damage and irradiation induced pore swell . Though preparing fully dense bulk tungsten is challenging, it is revealed in this presentation that laser triggered local melting by Selective Laser Melting (SLM) is a feasible way for achieving full density on tungsten surface by completely eliminating pores in the melted layer with a thickness of about 50 um. The subsequent rapid solidification of melt yields the grain orientation and 2D texturing that favors the improvement of the directional thermal conductivity thus the increase of damage tolerance of tungsten. It is found that the elimination of surface pores and the formation of surface 2D texture are determined by the maximum temperature and temperature gradient achieved on surface, respectively, which in turn can be manipulated by adjusting the laser power and scanning strategy.

Id 665

Abstract Final Nr. P4.160

Problems encountered during selective laser melting of tungsten

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Selective laser melting (SLM) has been proposed as a novel approach for preparing bulk tungsten with 3D textured microstructures that may demonstrate improved damage tolerance demanded for ITER applications. Building tungsten from powder granules in this way of droplet by droplet in three dimensions in an inert atmosphere is challenged by many problems that are hardly encountered during conventional materials processes, e.g. rapid oxidation when the oxygen partial pressure is above a critical level, crack formation due to the heterogeneous solidification shrinkage and the high residual thermal stress established between the adjacent layers, and inhomogeneous density distribution and the formation of large voids due to the imperfect packing of powder granules. Part of these problems is related to the powder properties, especially the oxygen content and their followability; part is related to the laser processing parameters, i.e. the applied laser power, the laser focusing size and the laser scanning strategies; the rest is determined by the capabilities of the selective laser melting facility, for example the achievable lowest oxygen partial pressure, the highest laser powder, the smallest laser focusing size and the fastest laser scanning speed etc. In this presentation efforts will be made to reveal the origins of these problems and to explore the methodologies for solving them. It will be demonstrated that joint efforts by software developers, machine builders and materials developers are necessary in order to solve these problems and to take the full advantages of this additive manufacturing process based on CAD/CAM principle.

Id 681

Abstract Final Nr. P4.161

Microstructural and mechanical characterization of Cu-0.8wt%Y

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Plasma facing components of the future fusion reactors require as effective heat dissipation that only Cu and Cu-based alloys can provide. Thus, the softening temperature and mechanical strength, along with the thermal conductivity and oxidation resistance at elevated temperature, are the key issues of concern in developing Cu alloys for in-vessel heat sinks. Alumina dispersion strengthened Cu as CuAl25, and precipitation hardened CuCrZr, are presently reference materials for these applications. The lack of thermal stability and low recrystallization temperature of the CuCrZr alloys, and the possibility of increasing the softening temperature and oxidation resistance via dispersion of stable oxygen scavengers, point out dispersion strengthening as the most immediate method for developing Cu with the demanded properties. Moreover, it would be expected that the fine dispersion of particles act as powerful sinks for irradiation induced defects increasing radiation resistance. Dispersion strengthened Cu-0.8wt%Y has been produced by consolidation of atomized pre-alloyed powder obtained from a vacuum induced melted ingot, and the microstructure characterized by X-ray diffraction and scanning electron microscopy (SEM). The powder particles exhibited hypoeutectic structure consisting of small Cu grains (< 30 micrometers) separated by thin layers with a eutectic structure. The particle size distribution and the oxygen content of the powder have also been determined. After consolidation by hot isostatic pressing at 850 ºC and 172 MPa for 2 h the alloy exhibited a dual grain structure of submicron sized grains and large grains with sizes of 20 micrometers. SEM analyses revealed the transformation of hypoeutectic structure into a uniform dispersion of Y-rich particles within the grains and along the grain boundaries. The tensile properties of the consolidated alloy over the temperature range 22 – 500 ºC, and microhardness measurements, will be presented.

Id 419

Abstract Final Nr. P4.162

Microstructural and mechanical characteristics of ITER-grade CuCrZr processed by equal channel angular pressing

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Equal channel extrusion pressing (ECAP) is among the severe plastic deformation processes applied to improve the properties of the material by introducing drastic changes on the grain structure and texture. The precipitation hardened CuCrZr alloy is a promising heat sink and functional material for components of the first wall, divertor and heating systems in ITER. In this study, the microstructure and mechanical properties of a Cu-0.65 wt.%Cr-0.08 wt.%Zr alloy processed by ECAP have been studied. The material has been ECAP deformed up to four passes at 400 ºC via route C; which consists in rotating the sample 180º around its longitudinal axis before inserting it in the die for the subsequent ECAP pass. It is observed that grain refinement and redistribution of the precipitates have been achieved after four ECAP passes. The microhardness and the Young's modulus increase from 178 HV and 44.4 GPa to 203 HV and 73.4 GPa, respectively. Mechanical tensile tests have been performed at room temperature in both flow and transverse planes. It is observed that the properties were improved with ECAP deformation. Further tensile tests at different temperatures are being performed to elucidate the deformation mechanism.

Id 649

Abstract Final Nr. P4.163

Magnetic properties qualification of European Reduced Activation of Ferritic-Martensitic steel ASTURFER®

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Local plasma rippling effects in ITER, at TBM Ports (2, 16, 18) depends on TBM masses and on magnetic properties of the material. Use of ferromagnetic structural material validated data are needed for Vacuum Vessel DEMO design. Even though magnetic characteristics are inherent part of qualification of structural low activation fusion reactor steels in electronuclear devices, experimental data on magnetic properties are scarce. Low activation (FM) Ferritic-Martensitic steel ASTURFER® has been developed under the Spanish National Fusion Technology Programme. ASTURFER® reproduces the material specification: (1) reduced-activation composition, (2) martensitic microstructure, (3) grain size, (4) d-ferrite precipitates and (5) thermo-mechanical properties; improving some of the characteristics of the reference material (EUROFER). This paper provides magnetic characterization data of low activation ferritic-martensitic steels ASTURFER® and EUROFER® samples within the material´s operational temperature design window. Magnetic tests at 400 and 450 C to obtain the hysteresis loop of Asturfer® and Eurofer steels have been carried out. Magnetization at saturation lies between 2.2 - 2.4 Tesla. The coercivities are small: for sample AFB2 it is of 2 mT at 673 K and 0.5 mT at 723 K; and for EF2 is even lower: 1 mT at 673 K and under measurement resolution (2 mT) at 723 K. The magnetic permeabilities are about: 1.1 x 10-5 for AF2B (at both temperatures) and 1.13 x 10-5 for EF2. The magnetic remanence is very low for both of them: 3 Tesla at 673 K and 0.005 Tesla at 723 K for Asturfer® and about 0.01 Tesla at both temperatures for Eurofer. Magnetic losses requiring dynamic measurements are not assessed. As a general feature, no significant differences between hysteresis cycles at both temperatures can be noticed and just a light decrease in the maximum M as expectable. The cyclic behavior of both two steels are also very similar.

Id 991

Abstract Final Nr. P4.164

Results of the RAMI analyses performed for the IFMIF accelerator facility in the engineering design phase.

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The planned International Fusion Materials Irradiation Facility (IFMIF) has the mission to test and qualify materials for future fusion reactors. IFMIF will employ the deuteronHlithium stripping reaction to irradiate the test samples with a highHenergy neutron flux. IFMIF will consist mainly of two linear deuteron accelerators, a liquid lithium loop and a test cell. The IFMIF accelerator facility is composed of two independent linear accelerators, each of which produces a 40 MeV, 125 mA deuteron beam in a continuous wave mode at 175 MHz. These beam characteristics pose several unprecedented challenges: the highest beam intensity, the highest space charge, the highest beam power and the longest RFQ (Radio Frequency Quadrupole). As a result of these challenges, many design characteristics are counter to highH availability performance: the design is reluctant to accept failures, machine protection systems are likely to stop the beam undesirably, cryogenic components require long periods for maintenance, and activation of components complicates maintenance activities. These design difficulties, together with the high availability requirements and the demanding scheduled operational periods, make RAMI analysis an essential tool in the engineering design phase. An iterative process was followed to match IFMIF design and availability studies. These iterations made it possible to include recommendations and design change proposals coming from the RAMI analyses into the accelerator reference design. Three different approaches were carried out in the iterative process. First, a comparison with other similar facilities was performed. Second, an individual fault tree analysis was developed for each system of the accelerator. Finally, a Monte Carlo simulation was performed for the whole accelerator facility considering synergies between systems. These approaches make it possible to go from detailed hardware availability analyses to global accelerator performance, to identify weak design points, and to propose design alternatives as well as foresee IFMIF performance, maintenance and operation characteristics. This paper presents a summary of the analyses done, the results obtained and the conclusions drawn. The consequences of the incorporation of the RAMI studies in the IFMIF design are described and the main outcomes of these analyses are shown.

Id 1029

Abstract Final Nr. P4.165

Synergistic effects of ELMs and steady state H and H/He irradiation on tungsten in plasma facing components.

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(3) Max Planck Institute for Plasma Physics, Garching, Germany

Plasma facing components endure a harsh combination of heat loads, neutron radiation and particle fluxes. ITER will therefore use actively cooled components which are covered with tungsten at the divertor. While the effects from different loading types are known, more research is needed for combined loading conditions. To study the effects of hydrogen/helium and hydrogen exposure on the behaviour of tungsten during edge localized modes (ELMs), particle loading in GLADIS and subsequent ELM-like loading in JUDITH 1 was performed. Post-mortem analysis was done by SEM, laser induced desorption (LID) and profilometry. Actively cooled mock-ups were manufactured with polished double forged tungsten samples (5x10 mm² surface, 5/10/15 mm height) brazed onto CuCrZr structures. Two mock-ups were loaded in GLADIS with pure H and a mixture of H/He particles, generated from a 30 kV ion source. The beam has a peak flux of 3.7x1021 particles/m²s, causing 2x1025 particles/m² fluence. After GLADIS exposure, all samples exhibited an altered surface morphology, while some had a porous coral-like surface structure. Thereby, the roughness increased up to Ra 0.43 µm, from initially Ra of 0.08 µm. Using LID, the He and H-content was determined in the near surface layer, concluding that helium does not influence near-surface H-content. These GLADIS loaded samples were exposed to 100 ELM-like pulses of 1 ms in JUDITH 1 at RT and 400°C with a power density of 190 and 380 MW/m². Post-mortem analysis showed no thermal shock resistance deterioration in comparison with polished material. On the contrary, for some test conditions, reference specimens roughened or cracked while the H and H/He exposed samples had no damage after JUDITH-exposure. Furthermore, analysis suggests that the surface morphology might become more regular. These investigations show that there is no additional ELM-damage inflicted on the H and H/He samples than on reference samples. Certain combinations of loading conditions could even result in a lower accumulated damage.

Id 412

Abstract Final Nr. P4.166

Thermomechanical properties of electron beam modified elastomer/polypropylene composites: insulation materials for nuclear power plant facilities

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Modification of thermoplastic and elastomeric polymers by radiation-chemically methods results in the formation of three-dimensionally cross-linked macromolecular structure, determined by a number of unique properties, compared to unmodified materials: an improved performance at reduced and elevated temperatures, increased thermal, chemical, and mechanical resistance, etc. significant exploitation properties. Creation of such composites is beneficial economically in the application of multifunctional thermoshrinkable polymer materials (TSM) that have a “form-memory” effect at elevated temperatures. The investigation of the rubber type TSM based on elastomeric severs is attributed to their application in the national economy, such as engineering facilities in nuclear and industrial applications, which are increasingly intensified worldwide. Some of the possible applications are insulation and coating materials, protection coatings against corrosion of iron constructions. Ethylene propylene diene (EPDM) and nitrile butadiene (NBR) blend composites with small content (30 wt.%) of polypropylene (PP) containing acrylate cross-linking agents were irradiated with accelerated electrons up to 150 kGy. The dependence of tensile properties at 20…150oC temperature, gel fraction, as well as the changes of thermoshrinkage characteristics of oriented up to 100% blends have been investigated. The strength properties (elastic modulus and tensile stress) of composites modified with cross-linking promoters increase significantly already at relatively low ionizing radiation doses (50-100 kGy). It has been stated that radiation-chemically modified compositions modified with acrylate cross-linking promoters can be successfully used as thermoshrinkable materials, as indicated by the relatively high content of gel fraction (40-85%) as well as the adjustable thermorelaxation stresses equal up to 0.6 MPa) and the residual shrinkage stresses equal up to 3.5 MPa).

Id 1008

Abstract Final Nr. P4.167

Brazing development and interfacial metallurgy study of tungsten to copper

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This paper presents the procedures developed for the successful brazing of tungsten to oxygen free high conductivity (OFHC) copper using a fusion appropriate gold based brazing alloy, Orobraze 890 (Au80Cu20). The objectives were to develop preparation techniques and brazing procedures in order to produce a repeatable butt joint for tungsten to copper, and to further the understanding of the effects that different preparation techniques and jigging arrangements have on the quality of the braze. The parent materials were examined pre-braze to determine surface roughness and the extent of microcracking and chipping. These findings are correlated with the results of the assessment on the vacuum brazed specimens. The specimens were sectioned to assess the wetted area of brazing alloy. Optical and scanning electron microscopy were utilised to determine microstructural features around the interface. Nano-indentation was performed on sectioned joints to examine microstructural features, joint quality and wetted area. Nano-indentation coupled with EDX elemental analysis across the brazed interface was used to determine any variation in the diffusion of brazing alloy, copper and tungsten for each brazing set up.

Id 1002

Abstract Final Nr. P4.168

The microstructure observations of W-Cu dissimilar brazed joint using Au80Cu20 braze alloy

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This paper summarised and discussed the observations of microstructure properties in the brazed joint created between pure tungsten and oxygen free high conductivity (OFHC) copper. The challenges in developing brazed joints of tungsten were discussed. The objectives were to develop a repeatable butt joint and to further understanding of the metallurgy of the interfacial regions. A novel gold based brazing alloy Au80Cu20, in wt%, was used for brazing. Butt type brazed joint specimens were created in vacuum by furnace and induction brazing. Microstructure properties derived by both brazing methods were compared and discussed. Optical and scanning electron microscopy was performed to assess microstructural features, joint quality and analyze the brazed interface between brazed filler and parent materials. No elemental diffusions were found between W and AuCu. Otherwise, a smooth elemental transition was achieved between Cu and AuCu. Nano-indentation was performed at brazed joint regions to generate mechanical properties and the results were correlated with EDS elemental interface to examine the effects of between parent materials and brazing alloy.

Id 1006

Abstract Final Nr. P4.169

Experimental Determination of Solubility Values for Hydrogen Isotopes in Eutectic Pb-Li

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Hydrogen isotopes solubility in eutectic lithium-lead alloys is really important for the design of breeding blanket components that use this breeding material. The determination of the magnitude and kinetics of the tritium flux from the blanket to the helium cooling loop, along with the design of future tritium extraction systems of the breeding alloy or the He coolant purification system, will be defined on basic transport parameters such as solubility. The unacceptable scattering of Sieverts' constant values in the historical measurements given by different experimental techniques, suggests that this is a very important and unresolved issue. In this work, it has been experimentally evaluated, using absorption and desorption techniques. The different measurement campaigns have been carried out in the temperature range from 523 K to 922 K and in the pressure range from 1 Pa to 105 Pa. This paper describes the work carried out in the preparation of the facility, the theoretical model developed to process the different results obtained by means of absorption and desorption runs. Final results obtained during several campaigns of measurements are provided. The obtained values of Hydrogen solubility through the different campaigns show a similar value for the Sieverts' constant, and therefore, a very little value for the activation energy in the solution process. However, this solubility value is higher than the historic Isovolumetric Desorption Experimental (IDE) measurements and higher than computed theoretical values of solubility.

Id 874

Abstract Final Nr. P4.170

Influence of the P content on the transport parameters of hydrogen in Fe alloys

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In this work, the hydrogen transport parameters of permeability (ϕ), diffusivity (D) and Sieverts´ constant (KS) were experimentally measured in four Fe alloys supplied by the European Fusion Development Agreement (EFDA), by means of the gas evolution permeation technique. The samples had controlled chemical alloying element contents and microstructure. The experimental temperature range explored was from 423 K to 823 K and the high purity hydrogen loading pressures from 103 Pa to 1.5x105 Pa. The main objective of this work was to determine the influence of the P content of the alloy in the transport parameters of hydrogen. Two of the samples, pure Fe and FeC, contained negligible quantities of P (less than 5 ppm in weight), whereas the other two, FeP and FeCP, had the same metallurgical composition as their corresponding pair, with the only difference in the phosphorus content (89 ppm in weight and 88 ppm in weight, respectively). The experimental permeation results were analysed using a non-linear least square fitting. The final resulting values of the aforementioned transport parameters were paired off in order to determine the effect of the P content: pure Fe versus FeP and FeC versus FeCP. We observed that the permeability obtained for all the samples follows an Arrhenius law in each case. In general terms, the increase of the P content in the alloy leads to smaller values of the permeability showing a decrease in the permeation activation energy. Regarding diffusivity and Sieverts´ constant, trapping effects have been observed for the alloys containing P. This phenomenon was detected at temperatures below 473 K. According to the results, the influence of the metallurgical composition of P in Fe alloys in the transport parameters of hydrogen is studied together with the synergistic effects caused by the presence of C.

Id 619

Abstract Final Nr. P4.171

Tensile and impact properties of CLAM steels as ITER-TBM-China candidate structural material after 1~3 dpa neutron irradiation

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The Reduced Activation Ferritic/Martensitic steels (RAFMs) are the primary candidate structural materials for the fusion DEMO and reactors in the future. The structural materials for future fusion reactors will undergo high displacement damage by intense fluence of high energy neutrons. So irradiation experiments on candidate fusion structural material are widely carried out to investigate the irradiation effects. Study on China Low Activation Martensitic (CLAM) steel was started ten years ago by FDS Team under wide collaboration with many institutes and universities in China. And the CLAM steel is chosen as the candidate structural material of the FDS series fusion reactor designs and ITER Test Blanket Module (TBM) of China (ITER-TBM-China). To investigate the irradiation effects on CLAM steel, irradiation experiments are being carried out under wide collaboration in China and overseas. In this paper, to investigate the irradiation-induced hardening and embrittlement on CLAM, the tensile and impact properties of CLAM (HEAT 0912A and 0603A) had been tested before and after neutron irradiation to 1.25 and 2.5 dpa in High Flux Engineering Test Reactor (HFETR) in China. The tensile test showed that ultimate strength and yield strength increased about 30MPa when tested at 200C and 300C. With the increase of test temperature, irradiation-induced hardening and elongation had been decreased. Impact test result showed that DBTT shift was about 10C. And comparison of hardening for CLAM and other RAFMs under similar irradiation conditions were introduced and analyzed.

Id 927

Abstract Final Nr. P4.172

The influence of helium-vacancy clusters on the critical resolved shear stress of Fe-Cr alloys

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Reduced Activation Ferritic/Martensitic (RAFM) steels are considered as the primary candidate structural materials for the Test Blanket Module (TBM) in International Thermonuclear Experimental Reactor (ITER), whose basic composition is Fe-Cr alloy. In fusion reactor enviroment, specially 14MeV neutron, massive helium produced by (n,α) reaction become a key factor to degrade mechanical performance of RAFM. In this paper, the effect of helium-vacancy cluster on the mobility of a/2<111>{110} edge dislocation in Fe-Cr alloy is investigated. A possible influence of Cr on the mobility of a/2<111>{110} edge is evaluated. The molecular dynamics (MD) methods and interatomic potentials for the Fe-Cr-He system recently developed by Juslin et al. are applied in this study. The interaction between helium-vacancy cluster and a/2<111>{110} edge dislocation in Fe-Cr alloy with different proportion of He/V and Fe/Cr by classical MD code LAMMPS. Then critical resolved shear stress for dislocation glide is calculated from the results of modeling. The results indicates that critical resolved shear stress for dislocation glide significantly rely on the size of helium-vacancy cluster and He/V ratios, but slightly depend on proportion of Fe and Cr.

Id 946

Abstract Final Nr. P4.173

Elevated temperature performance assessment of CFETR candidated structure materials

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Chinese Fusion Engineering Test Reactor (CFETR) would be a good fusion neutron source for future fusion reactor material test. The major missions of this ITER-like superconducting tokamak reactor are to generate 50-200MW fusion power, and to realize 0.3-0.5 of duty time for burning plasma with long plus operation or steady state operation, at the same time to achieve the tritium breeder ratio(TBR) for blanket not less than 1.2 to ensure the self sufficiency.As a risk mitigation strategy, three blanket concepts are under development and evaluation in parallel. For the high heat flux in the first wall (FW) and high power blanket concept design, high temperature performance become a big challenge for materials. According to present blanket designs, structure material would be in service at high temperature under low dose irradiation with long duty time. And strength, creep, low cycle fatigue and creep-fatigue interaction would be very important for materials selection of high temperature structural components. Normally, reduced activation ferritic martensitic (RAFM) steel was considered as baseline material. But internal industrial matured austentic and ferritic/martensitic steels was assessed to be an alternative option from the practical point of view in present research, in which materials datebase, stress analysis and lifetime assessment were emphasized. Besides, oxide dispersion strengthened (ODS) steel would be a better option in the near future in the case of good R&D development.

Id 948

Abstract Final Nr. P4.174

Two-phase MHD energy conversión from buoyancy-driven flows of liquid metal coolant

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The present consensus for generating electrical power from a tokamak is to use a well-established energy conversion method: the conversion of heat to electricity through gas or steam turbines. The overall objective of our research is to compare alternative methods of conversion. This work focuses upon the PPCS-C, a plant design considered to be realistically near-term in which the blanket modules are cooled by both helium and the Pb-17Li liquid metal breeder. Consideration of the high-enthalpy liquid metal has led to the study of two-phase MHD generators, in which an added gas phase reduces the detrimental effects of high pressure loss associated with single-phase liquid metal MHD power generation. In the proposed concept most of the helium is at lower temperature and is most efficiently expanded in a Brayton cycle, but a small fraction is injected into the liquid metal after it has left the blanket, resulting in direct-contact heat exchange. The two-phase mixture subsequently flows around a vertical loop, first passing through the MHD generator and then rising in an upcomer to a reservoir, where the phases are separated. The helium returns to the Brayton cycle, whilst the liquid metal rejects excess heat by indirect exchange to the Brayton cycle before flowing in a downcomer to the blanket, thus completing the loop. This configuration generates small amounts of power by MHD, whilst simultaneously using buoyancy to circulate the liquid metal through the blanket, thereby eliminating a pump. The power cycle configuration therefore involves a Brayton cycle with liquid metal two-phase MHD topping and Rankine cycle bottoming. Several studies have been undertaken and a computer model developed to predict the net power output of the fusion plant and the subsequent overall plant efficiency. Small gains over a conventional configuration appear feasible.

Id 1037

Abstract Final Nr. P4.175

numerical research on factors influencing dust migration during a loss of vacuum accident

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During a loss of vacuum accident (LOVA), dust migration is related to the safe operation of ITER device. The migration of dust brings not only the risks of hydrogen explosion and dust explosion, but also the risk of radioactive hazards. Therefore, it is necessary to study the factors influencing dust migration during a LOVA so as to provide theoretical basis for the elimination or mitigation of risks. Computational fluid dynamics (CFD) code CFX was used to simulate dust migration during a LOVA in the experimental model. Wall temperature and obstacles were considered to be the factors influencing dust migration. Through setting different wall temperatures and changing the existence of obstacles, the effects of these conditions were studied. In terms of dust properties, particle diameter and density were considered to be the factors influencing dust migration. Through comparing different distributions of carbon dust and lithium dust, the effects of dust properties were studied. The simulation results show that during the LOVA, the wall temperature has some effects on dust migration. When the wall temperature increases, the migration of particles is restricted a little. The existing of obstacles can limit the migration of dust obviously. Both the results agree well with the tendency of experimental results. For the same kind of dust with different diameters, it is found that the distribution of dust is more similar to the velocity field of gas when the dust diameter is smaller. For different kinds of dust with the same diameter, it is found that the distribution of dust is more similar to the velocity field of gas when the density is smaller. The results above is consistent with the theory of stokes number. This research is funded by National Natural Science Foundation of China (No. 11375116) and Chinese National Fusion Project for ITER (2013GB114005, 2014GB122000).

Id 889

Abstract Final Nr. P4.177

Validation of SEACAB methodology with Frascati (FNG) photon dose measurements

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In the operation of the International Thermonuclear Experimental Reactor (ITER) the correct estimation of the gamma dose rate produced from the structural materials after shut down is one of the important safety parameter for hands-on maintenance. SEACAB, a rigorous 2-step (R2S) computational method has been developed for the calculation of residual dose in 3-D geometry with the use of the MCNP5 and of the ACAB (ACtivation ABacus) inventory code. The method is very efficient in hardware requirements being essentially modular. Starting from a single MCNP5 run permits a progressive improvement in the spatial detail of the material layers for the activation calculation and obtains separated photon source distributions for the isotopes contributing to the photon dose. Validation of the system has been achieved through the computational analysis of the Frascati Neutron Generator (FNG) experiment of residual dose and photon flux. A mock-up of ITER shielding blanket made of stainless steel and water-like material with a channel and cavity inside the bulk shield was irradiated with 14MeV neutrons induced by D-T reaction. The gamma dose rate measurements were performed in the cavity after shutdown as a function of decay times 1day, 7days, 15days, 30 days and 60days. The calculations are performed with FENDL 2.1 library for neutron transport and EAF2007 library for activation. A full 3D detail mock-up geometry has been created and both the analogous calculation and a weight windows variance reduction technique are applied to obtain less than 2% statistical uncertainty on total neutron flux and less than 1% on photon flux.

Id 149

Abstract Final Nr. P4.178

Preliminary experimental study of liquid lithium water interaction

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Liquid lithium is the best candidate for a material with low Z and low activation, and is one of the important choices for plasma facing materials in magnetic fusion devices. But lithium has active chemical properties. Liquid lithium reacts violently with water under loss of coolant accident. The release of large heat and hydrogen could result in the dramatic increase of temperature and pressure. Considering the loss of vacuum accident, hydrogen explosion may occur. The lithium-water explosion has large effect on the safety of fusion devices, and is an important content for safety assessment. As a preliminary investigation of liquid lithium water interaction, the test facility has been built and experiments have been conducted under different conditions. The initial temperature of lithium droplet ranged from 200º to 600º and water temperature was varied between 25º and 90º. Lithium droplets were released into the test section with excess water. The shape of lithium droplet and steam generated around the lithium were observed by the high speed camera. At the same time, the pressure and temperature evolutions in the vessel were recorded during the violent interactions. The preliminary experimental results indicate that the initial temperature of lithium and water have an effect on the violence of liquid lithium water interaction. This research is funded by Chinese National Fusion Project for ITER (2013GB114005, 2014GB122000) and National Natural Science Foundation of China (No. 11375116).

Id 890

Abstract Final Nr. P4.179

Behavior Analysis and Computer Simulation of Activated Corrosion Products in Fusion Reactors

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Activated corrosion products (ACPs) are the dominated source of radiation hazard in fusion reactors, and influence the occupational radiation exposure directly. Therefore, it is necessary to carried out the source term analysis of ACPs in the primary cooling loops of fusion reactors. In this work, a code named CATE was developed for Corrosion Activation Transport Evaluation, which consisted of balance equations of nuclides, the matched numerical method and a specialized database. Firstly, the behavior of ACPs in the cooling loop, including generation, transport, deposition and so on, was investigated. As a result, the balance equations of nuclides were established, and were simplified with some reasonable assumptions. Then the fourth-order Runge-Kutta method was adopted to solve the equations. Meanwhile, based on the selected materials, water chemistry, thermal-hydraulic and radiation conditions in the cooling loop, a database was set up to provide the constants appearing in the equations, such as corrosion rate, dissolution rate, deposition rate, activation cross section, and decay constant. Finally, the cooling loop of ITER were simulated using CATE, and the radioactivity of ACPs in the coolant and on the pipe surface were calculated respectively.

Id 337

Abstract Final Nr. P4.180

Validation of cryostat-vacuum vessel aerosol tracing and FSI using MELCOR and COMSOL with small scale experiment

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One of the most important values in the nuclear fusion system is how much the radioactive material would be leak out from the TOKAMAK where the real fusion take places. Tracing this radioactive material is indispensable process to build nuclear fusion plant or research reactor like ITER. If severe accident like penetration between vacuum vessel and outer environment occur, this radioactive substance will harm the envrionment around. To predict this severe acident, until now, there were many kinds of validation of this radio active material leakage amount by various institution. Especially modified MELCOR version for ITER is used in validation of this experiments. For example, EVITA (Europe Vacuum Impingement Test Apparatus) validates the amount of condenced air, ice or water in the impingement through the vacuum vessel and cryostat to environemnt. And also research of aerosol across the plant is studied. In this research, using MELCOR and COMSOL, experiment of movement of aerosol in vacuum vessel to air and structure interaction would be conducted and validated to measure the burden of vacuum vessel and amount of radioactive material in the simulation of impingement test like severe accident in ITER system. With this research, the validated value of stress on structure and amount of aerosol would be helpfull to investigate how much nuclear fusion plant or nuclear fusion research reactor is safe.

Id 899

Abstract Final Nr. P4.181

Study of neutron and shutdown dose rate cross-talk from lower to equatorial ports in ITER

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One of the main issues arising due to activation of ITER port systems is that of occupational radiation exposure (ORE) during planned in-situ maintenance activities in the port inter-space (PI) area. ITER ports and port plugs (PP) become activated by neutron irradiation during plasma operation and the so-called "shutdown" dose rate field (SDR) appears as irradiated materials decay. Project design target for routine maintenance at PI exists: SDR below 100 uSv/hr after 12 days. This is particularly challenging for equatorial ports (EP) due, among others, to the background neutron radiation/activation produced from/by neighbouring components (a.k.a. "cross-talk"). The study reported here is focused on the evaluation of shielding options to reduce neutron and SDR cross-talk from the LP to the EP in the particular case of a torus cryopump (TCP) lower port (CLP) and a generic diagnostics equatorial port (DEP). Scoping neutron field analyses were performed of the reference situation plus up to 34 shielding variants based on combinations of six shields located in different parts of the LP, TCP, VV, cryostat and equatorial PI. The study on the reference case showed that about 50% of the neutron flux in the equatorial PI gets there through the CLP. The most effective shielding variants appeared to be those located inside the LP, near the neutron source, achieving up to a 40% reduction in the flux at EP PI; however these are engineeringly challenging and compromise TCP performance. Performance of other, more manageable and neutral options is poorer, typically <15%. Full SDR field computations are currently being performed on a sub-set of the shielding variants. The design of the shielding options is subject to further analysis to be taken into account on the basis of engineering feasibility, performance neutrality, and effectiveness in reducing the neutron flux at the equatorial PI. This work has been funded with support from Fusion for Energy. This publication reflects the views only of the author, and Fusion for Energy cannot be held responsible for any use which may be made of the information contained herein.

Id 1014

Abstract Final Nr. P4.182

Optimization of a steel/water based shield for European TBMs HCLL and HCPB and analysis of the resulting Shutdown Dose Rates

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The European Test Blanket Modules (TBM) are based in two approaches: Helium Cooled Lithium Lead (HCLL) and Helium Cooled Pebble Bed (HCPB). The TBMs will be attached to the corresponding TBM shields, and at the same time, they will all be inserted into the Port Plug Frame (PPF). These systems will be hosted in ITER Equatorial Port (EP) #16. Shutdown dose rate (SDDR) at EP Port Interspace (PI) is currently one of the main design drivers for these components. Project requirements exist to meet the regulatory limit for Operational Radiation Exposure: 100 uSv/hr after 12 days. In this study it is presented a justification for a TBM shield made of steel and water, plus a SDR analysis at PI considering this TBM shield. In the first part of the study, it is presented an optimization process of the TBM shield based on neutron flux attenuation factor performance and activation response for different steel/water distributions inside the shield (more than 20 cases). As a result, a shield divided into three shielding bodies separated by two water tanks, hosting helium and Lithium-Lead pipes was reached. Each shielding body presents a steel/water ratio of 70%/30%, 70%/30% and 25%/75% respectively ordered from the plasma source. This solution offered an attenuation factor of 3•107. Then, a constructive solution was generated for this shield, considering water and steel layers, plus helium, LiPb and diagnostics pipes. In the second part, an MCNP model of the PPF, TBMs and TBM shield is developed and inserted into Blite v3 improved for EP analysis. Then, using the R2S-UNED system, a full characterization of the SDDR at PI after 12 days of cooling time is made: i) SDDR maps at PI, ii) identification of main decay sources contributing to the SDDR, and iii) identification of the main path followed by neutrons to generate the key decay sources. This work has been funded with support from Fusion for Energy. This publication reflects the views only of the author, and Fusion for Energy cannot be held responsible for any use which may be made of the information contained herein.

Id 1017

Abstract Final Nr. P4.183

Characterisation of metal combustion with DUST code.

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The code DUST is a CFD code developed by the Technical University of Cartagena (Spain) and the IRSN (France) with the objective to assess the dust explosion hazard in the vacuum vessel of ITER. Thus, DUST code permits the analysis of dust spatial distribution, remobilisation and entrainment, explosion, and combustion. Some assumptions such as particle incompressibility and negligible effect of pressure on the solid phase make the model quite appealing from the mathematical point of view, as the systems of equations that characterise the behaviour of the solid and gaseous phases are decoupled. In this work the model implemented in the code to characterise metal combustion is presented. In particular, the paper is focused on two combustion problems involving the analysis of reactive mixtures of multicomponent gases and multicomponent solids, namely H2/N2/O2/C and H2/N2/O2/W mixtures. The system of equations considered and finite volume approach are briefly presented. The closure relationships used are commented and special attention is paid to the reaction rate correlations used in the model. The numerical results are compared to those obtained experimentally at the IRSN/CNRS facility in Orleans. They are commented and some conclusions are finally drawn.

Id 761

Abstract Final Nr. P4.184

Design of a new experimental facility to reproduce LOVA and LOCA consequences on dust resuspension

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Dust Resuspension inside the Vacuum Vessel is one of the key security issues of the new-generation tokamak (like ITER or DEMO). It is well known that a fusion device generates dusts by energetic plasma-surface interactions, which cause a significant erosion of the vacuum vessel (VV) internal wall materials. Consequently, operators will have to manage several hundreds of kilograms of beryllium and tungsten dusts inside the VV. According to the reference categories, two main accidental situations lead to dusts re-suspension: Loss Of Vacuum Accidents (LOVA-air flow due to a rupture of a penetration line) and Loss Of Coolant Accidents (LOCA-fluid flashing due to a rupture of a coolant system pipe). The authors have gained a strong experience in the field of dust resuspension by virtue of the studies on STARDUST facility, whose limitations, however, prevent from completing further analysis. These are, in particular, a reduced field of view to track the dust with optical techniques [1,2], the impossibility to replicate a LOVA from the upper port as well as any kind of LOCA. To overcome these problems, several new layouts of the facility have been designed. Numerical simulations to test the mechanical resistance together with a deep analysis of advantages and limitations have been performed for each layout. The authors will present the proposals for the new facility, the numerical results of the simulations and a comparison between the layouts analysed. A new experimental facility will be then revealed to reproduce dust resuspension due to both LOVA and LOCA consequences. [1]Malizia A. et all (in press). Dust tracking techniques applied at STARDUST facility : first results. FUSION ENGINEERING AND DESIGN, doi: 10.1016/j.fusengdes.2014.01.014 [2]Gaudio P. et all, (2013). Shadowgraph Technique applied to STARDUST facility for dust tracking: first results . In: Proceedings ICFDT 2013 (ICFDT2013). INFN- Frascati, 25-27/11/2013

Id 248

Abstract Final Nr. P4.186

Design, fabrication and testing of ESS-Bilbao linear accelerator components

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The process of design and fabrication of several accelerator components (cavities and focusing magnets) is described from the conceptual design stage to the prototype fabrication. Aspects related to fabrication control and validation are emphasized. The rebunching cavities for the MEBT section of the linac are important components for assuring beam quality. The fabrication of a cavity prototype is described, along with electromagnetic measurements and metrology quality control. The analysis by numerical methods of the differences observed allows to optimize the fabrication process in order to obtain final cavities with accelerator conforming quality. Similar procedure is also carried out with focusing quadrupole magnets, using magnetic measurements and metrology as production quality assurance techniques.

Id 816

Abstract Final Nr. P4.187

Pulse duty management system for the injector commissioning of Linear IFMIF/EVEDA Prototype Accelerator

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On the Linear IFMIF/EVEDA Prototype Accelerator (LIPAc), the validation up to the energy of 9MeV deuteron beam with 125mA continuous currents is planned in Rokkasho, Aomori, Japan. The three phase of LIPAc commissioning is planned; the 1st phase is the injector commissioning (D+, 140 mA - 100 keV), the 2nd phase is the RFQ that connected after injector commissioning (125 mA - 5 MeV), and the final phase is whole LIPAc (with SRF Linac) commissioning (125 mA - 9 MeV). In fact, after the end of injector commissioning, the RFQ will be installed the downstream of the injector. The injector will contain the many radioactivated materials by deuteron generation. Therefore, to reduce the radiation exposure for personnel during RFQ installation work, the measurement of injector beam output time is indispensable. To measure the injector beam output time, “Pulse Duty Management (PDM)” system was designed and developed by JAEA. The PDM system measures the injector timing signal “Gate signal for RF pulse” which decides the beam output time of injector. The PDM system mainly consists of a Programmable Logic Controller (PLC) and a Gate Signal Counter (GSC) based on Field-Programmable Gate Array (FPGA). In this system, we designed the highly-reliable data transfer between PLC and GSC. In addition, two PDM systems work as dual configuration to also implement the higher reliability. In this paper, the technical design of PDM system for high reliability and the result of operation test are presented in details.

Id 931

Authors' index

Invited speakers

A

C

E

F

G

H

I

J

K

L

M

O

P

Q

R

S

T

U

V

Z