

27th IEEE Symposium on Fusion Engineering

Sunday 4 June 2017 - Thursday 8 June 2017

Marriott Shanghai City Center



Book of Abstracts

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M.POS: Poster Session M - Board: 62 / 5

Current Status and Progress on the Shield Blanket Design of CFETR Reactor

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The main function of the CFETR shield blanket (SB) system is to provide the neutron shielding capability for the in-vessel components and the external environment. The SB concept design has been carried out during the 2009-2011 campaign. The current design on the SB system is concentrated on the neutronics analysis, the SB modules design, and the mechanics analysis using finite element method (FEM). The neutronics analysis on the SB thickness estimation in radial direction and the neutron shielding performance on the shield materials is carried out. The two kinds of SB module structure are taken into account respectively, which are mainly for exploring the heat removal capability comparing for the SB cooling channel system. In addition, the mechanics analysis is made on the structure static stress and the electromagnetic (EM) force considering the core plasma disruption. This paper summarizes the SB components design activities and the progress at present status.

Eligible for student paper award?:

No

T.OA1: Diagnostics and Instrumentation I / 8

Novel multi-energy x-ray cameras for magnetically confined fusion plasmas

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A compact multi-energy soft x-ray (ME-SXR) camera has been developed for time, energy and space-resolved measurements of the soft-x-ray emissivity in magnetically confined fusion (MCF) plasmas. Multi-energy x-ray imaging provides a unique opportunity for measuring, simultaneously, a variety of important plasma properties (T_e , $n_e^2 Z_{eff}$, n_Z , ΔZ_{eff} and $n_{e,fast}$). Selecting an appropriate detector response eliminates the contamination introduced by the low- and high-energy line-emission from medium- to high-Z impurities facilitating temperature measurements in Ohmic and RF-heated scenarios (e.g. ICRH and LHCD) in agreement with conventional ECE and Thomson scattering systems. Impurity density measurements are also possible using the line-emission from medium- to high-Z impurities to separate background as well as transient levels of metal contributions. This novel imaging system developed at PPPL and tested first at Alcator C-Mod tokamak at MIT, combines the best features from both pulse-height-analysis (PHA) and multi-foil methods, and represents a very large improvement in throughput and spatial resolution thanks to present state-of-the-art pixelated PILATUS detectors with nearly 100k pixels. Being the first of its kind, this novel diagnostic will be used to resolve the impurity emission, study impurity transport and impurity-induced MHD and will become an essential part of a control algorithm coupled to physics and engineer actuators for minimizing impurity accumulation in tokamaks. This technique should be explored also as a burning plasma diagnostic in-view of its simplicity and robustness. Recent results from a detector sensitivity study including its response at high magnetic fields (up to 3.4 T and 3.0 T/s) and ITER-like neutron fluences (up to $10^{15} - 10^{16}$ n_{eq}/cm^2) will be presented.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 84 / 9

Lithium Evaporation System Design for the NSTX-Upgrade Fusion Device

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The National Spherical Torus eXperiment (NSTX) has recently undergone a major upgrade to NSTX-U at Princeton Plasma Physics Laboratory (PPPL). To improve plasma performance for NSTX-U, control of the influx of impurity gases and fuel recycling from plasma facing components (PFCs) are critical. On the NSTX fusion device, two lithium oven evaporators (LITERs) were mounted on the upper dome of the vacuum vessel to apply a thin layer of lithium coating downward onto lower divertor, which has resulted in effective deuterium retention and improved energy confinement time. For NSTX-U, it is desirable to have lithium directly coat the upper divertor for double-null plasma operation, and increase the coverage of PFCs in general with lithium. In this paper, the design of a new lithium evaporation system capable of coating NSTX-U PFCs in all directions will be reported. Porous stainless steel (SS) tubes will be used to hold the lithium. Lithium will be loaded into the pores of the SS tube as a liquid at elevated temperature within an argon glovebox. A vacuum heater will be inserted into the lithium-loaded SS tube to heat lithium to more than 600 C for lithium evaporation in the NSTX-U vacuum vessel. With significantly less thermal mass compared to the LITERs, the new lithium evaporator can be heated up and cooled down much faster. The time needed to reach operating temperatures and unwanted lithium evaporation while at temperature will be drastically reduced. The new evaporation system setup and thermal modeling results of the temperature distribution achieved in the porous evaporator will be covered in detail. Safety design considerations and analysis regarding lithium handling during operation will also be included.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 57 / 11

NEUTRONIC SHIELDING DESIGN OF THE ITER EC UPPER LAUNCHER

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In four of the upper ports of ITER, Electron Cyclotron launchers will be installed for heating and plasma stabilization. The launchers are designed as stainless steel casks (so-called port plugs), accommodating microwave mirrors and waveguides with the capability to inject up to 24 MW total microwave power into the plasma.

The inner volume of the port plugs represents a relatively open structure which is unavoidable since the propagation of the microwave beams shall not interfere with any structural components. Thus it is essential to fill all remaining volumes with shielding components to guarantee the compliance of the launchers with the neutronic design requirements.

That is why the EC upper launchers will be armed with three particular shielding elements, of which the first one is installed into the upper area of the plasma-facing Blanket Shield Module, the second one in the front area of the launcher main structure and the third one in its rear part. All shielding components must be equipped with suitable internal cooling structures regarding volumetric heat dissipation, acceptable pressure drop and proper steel/water ratio for optimum shielding performance.

This paper outlines the general design of the shielding components, including mechanical structure, cooling layout and integration, interfaces with the MW-components, manufacturing, installation and maintenance aspects. Also analyses to prove mechanical integrity, thermo-hydraulic behavior and shielding capability will be presented.

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.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 63 / 13

Concept Design of GDT-Based Fusion Neutron Source for Improving the Q with High Field Neutral Beam Injection

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Gas Dynamic Trap (GDT) is very attractive as a kind of fusion neutron source for testing fusion material and component as well as driving transmutation reactor due to its linear and compact structure, easiness of construction and maintenance, relatively low cost and tritium consumption. These years, the conceptual designs of GDT-based neutron source for above two purposes, named FDS-GDT, have been proposed as candidate of fusion neutron source by Institute of Nuclear Energy Safety Technology (CAS) • FDS Team in China, which focus on fusion safety and fusion nuclear science and technology research. However, the fusion energy gains (Q) in current international designs are still far lower than one, even about 0.05.

In order to improve the Q and reduce the technologies requirement of magnet and neutral beam injection (NBI) for GDT-based fusion neutron source, a new method was proposed with high field neutral beam injection (HFNBI) for substituting the conventional method that the neutral beams are obliquely injected at middle plane of GDT where the field is minimal. This method will benefit for confining higher density of fast ions at turning point in the zone with higher magnetic field, as well as getting higher mirror ratio by reducing mid-plane field rather than increasing the mirror field. In this situation, the collision scattering loss of fast ions with higher density will be critical and change its confinement performance, power balance and particles balance.

Two optimal designs of GDT-based fusion neutron source was proposed with HFNBI by using updated calculation model and based on SYSCODE. One is for improving Q to 0.5, about 10 folds comparing to conventional design scheme, and the fusion power is 18MW. The other is for reducing the NBI power and mirror field to enabling level, such as 10MW and 10T respectively, and the fusion power is 2MW and Q is 0.2.

Eligible for student paper award?:

No

M.OP3: Next Step Devices, DEMO, Power Plants / 15

Preliminary Research on Reliability Index System of Fusion Power Plant

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Nuclear fusion is one of the most promising options for generating large amounts of carbon-free energy in the future. Since fusion energy is innovative and fusion facilities contain unique and expensive equipment, the reliability issue is very important from their efficiency perspective. The evaluation of reliability is an important part in the safety study of fusion reactor. And the system reliability index is the premise and the basis of reliability evaluation.

This paper aims to establish the reliability index system of fusion reactor. Firstly, the safety goals of fusion reactor were given in this paper. In this study, the safety goals were separated into quantitative safety goals and subsidiary numerical objectives. Quantitative safety goals are higher than the numerical objectives, which come from the two 0.1% risk limits defined by the United States Nuclear Regulatory Commission (USNRC). Subsidiary numerical objectives are actually developed under the quantitative goals and are more specific to the characteristics of fusion reactor. Secondly, the safety goals of fusion reactor were assigned to the components which performed safety functions. In the part of this study, the Probability Safety Assessment (PSA) was used to establish the risk models for fusion reactor. The PSA is an important method to evaluate the risk of system, which has rich experience applied in nuclear industry for fission power plants and other nuclear installations. Thirdly, the reliability index system was given based on the results of the risk analysis of fusion reactor.

The validation of reliability index system is still on study. The reliability index system is expected to be the basis and the reference for the reliability evaluation of fusion reactor and nuclear safety monitoring in future.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 26 / 17

Divertor heat flux study of H-mode with NBI in EAST

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H-mode with ELMs (Edge Localized modes) plasma regime is considered to be a preferable scenario in the future fusion devices as ITER. Heat loads on divertor during ELMs especially the type I ELMs is an important issue [1-2]. An infrared (IR)/visible endoscope system was built on the Experimental Advanced Superconducting Tokamak (EAST) in 2014. Based on the IR data in the experiment of 2014, the heat fluxes on the lower outer divertor were calculated with a code named DFLUX developed by ASIPP, aimed to provide reference for the H-mode operation of EAST [3]. Heat fluxes on lower outer divertor during ELMs without neutral beam injection (NBI) and with different NBI power were calculated and compared. The analyzed discharges were lower single null (LSN) divertor configuration discharges. In the case with lower-hybrid wave current drive (LHCD) only ($I_p \sim 400\text{kA}$, PLHCD $\sim 2\text{MW}$), ELM-free occurred after L-H transition accompanied by the increasing electron density (n_e). The peak heat fluxes on lower outer divertor during ELM-free were not more than $1\text{MW}/\text{m}^2$. Then ELMs occurred and n_e began to reduce and eventually lead to H-L transition. The peak heat flux on lower outer divertor during ELMs was not more than $2\text{MW}/\text{m}^2$ mostly. In the case LHCD combined with NBI ($I_p \sim 300\text{kA}$, PLHCD+PNBI $\sim 2\text{MW}$), type I ELMs occurred after L-H transition and n_e was still increasing. The peak heat flux on lower outer divertor during ELMs was more than $3\text{MW}/\text{m}^2$ or even $5\text{MW}/\text{m}^2$ and ELMs disappeared when NBI was turned off mostly. By comparing the heat flux profile of divertor target versus radius in ELM, it may be because that the heat flux profile in ELM with NBI was narrow. The peak and averaged heat flux on lower outer divertor during ELMs did not increase with the increase of PNBI. The work of this paper will provide reference for H-mode discharge with NBI of EAST.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 91 / 18

Time Synchronization Network for Poloidal Field Power Supply Control System Based on IEEE 1588

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This paper describes architecture and characteristics of time synchronization network of EAST poloidal field power supply control system. After analyzing the characteristics of IEEE 1588, the paper points out the advantages of IEEE 1588 and chooses it as the implementation protocol for time synchronization network in EAST poloidal field power supply control system. With this method, a master node with a hardware clock synchronizes time of the slave nodes on the private time network in each second, and the maximum offset time between the master and the slave is less than 50ns. This time synchronized method is evaluated and tested with its feasibility, and also met the requirements for the time synchronization network of EAST poloidal field power supply control system.

Eligible for student paper award?:

No

R.OP4: Stellarators / 19

HIDRA –A Stellarator for Materials Research

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In tokamaks the intimate relationship between plasma performance and materials used at the first wall and divertor are recognized but not well understood. More so, in stellarators, the role of plasma material interactions has not been as rigorously pursued since the optimization of the confinement properties has been the main focus. However with the next generation of stellarators and tokamaks coming on line or being designed the PMI issue needs to come more into focus. The tokamaks and stellarators offer different sets of PMI challenges that need to be addressed. However one of the main ones being long exposure of materials to the plasma. Thus it is important that a device exists where it is dedicated to materials research. The Hybrid Illinois Device for Research and Applications (HIDRA) is a toroidal plasma device at the University of Illinois and aims to be a dedicated, long pulsed stellarator/tokamak for plasma materials research. The vacuum vessel has a circular cross section and a major radius of $R = 0.72$ m and a minor radius $a = 0.19$ m, with a steady-state magnetic field < 0.5 T. A limiter can be used to reduce the plasma minor radius between 0.10 –0.15 m. HIDRA has the ability for long pulse steady state operation via the classical stellarator configuration and has an actual toroidal magnetic field, just like a tokamak. A pulsed capability during steady operation allows simulation of transient events. Initial plasmas will use 2.45 GHz magnetron heating up to 26 kW and should achieve $T_e \sim 20$ eV and $n_e \sim 1 \times 10^{18} \text{ m}^{-3}$. Though the plasma parameters are lower than that of larger devices like W7-X or EAST, the plasma and magnetic fields at the first wall are very close to those produced in HIDRA. These capabilities make HIDRA an ideal test bed for materials and PMI studies, for example liquid Li technology where the science and technology can be tested, understood and perfected first in preparation for a final design that would be installed on a larger device.

Eligible for student paper award?:

No

M.OA3: Inertial Fusion Engineering and Alternate Concepts / 21

Radiation Safety Design for the North Pole Neutron Time-of-Flight System at the NIF

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The National Ignition Facility (NIF) at Lawrence Livermore National Laboratory is the world's largest and most powerful laser system for inertial confinement fusion. During the ignition campaign, the NIF is expected to generate shots with varying fusion yield (up to 20 MJ or 7.1×10^{18} neutrons per shot). Neutron time of flight (nTOF) detectors are fielded in the NIF to measure neutron yield, ion temperature, and downscattering in the cold fuel for D-T implosions. A collimated nTOF line-of-sight (LOS) has been fielded near the Target Chamber (TC) North Pole to examine any possible anisotropy in the cold fuel. A fast scintillator is placed inside a diagnostics hut located above the roof of the Target Bay (TB). The scintillator is located at 21.6 m from the Target Chamber Center (TCC). The line-of-sight passes through the TC, the 60.96-cm-thick concrete 69'9" TB floor and the 76.2-cm-thick concrete TB roof. Radiation streaming through the LOS represented a potential radiation hazard if personnel were accidentally present on the 69'9" floor or on the top of the TB roof. The potential hazard at

these two locations is caused by radiation streaming through a 30.48-cm-diameter hole in the 69'9" concrete floor and the TB roof. Additional potential hazard to personnel present on the roof during a shot is caused by radiation scattering off the scintillator. The un-scattered radiation is eventually intercepted by a beam dump made of 45.72-cm-thick iron followed by 30.48-cm-thick concrete. The front surface of the beam dump is located at 23.85 m from TCC. The beam dump is designed to fully intercept the radiation and eliminate skyshine hazard due to neutrons passing through the LOS and interacting with the surrounding air.

A detailed MCNP model of the TB is used to estimate dose values at the previously identified locations of concern during shots. Before adding the new LOS, the area above the 69'9" floor wasn't normally accessed or swept before low yield shots ($< 1e16$ neutrons) due to expected low dose values. The MCNP simulation indicated that adding the LOS will result in a maximum effective dose of 2 mSv on the 69'9" floor during $1e16$ shot. Following construction of the LOS, the 69'9" TB floor is routinely swept and no access is allowed during all shots. Similar analysis showed that radiation scattering off the scintillator resulted in higher dose values inside the diagnostics hut. A maximum effective dose value of 3 mSv is expected outside the hut during a 20 MJ shot. Currently, no access is allowed inside the hut during all shots, and access control of the TB roof is required for shots with yield above $1e16$ neutrons. Finally, in addition to reducing the skyshine dose, the beam dump effectively eliminated any potential hazard to planes flying over the facility. In conclusion, contribution of the new North Pole nTOF system to dose outside the facility and near the site boundary is negligible.

*This work performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract DE-AC52-07NA27344.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 68 / 22

Research on Brazing technique of tungsten materials and reduced active ferritic–martensitic steels

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Co-authors: Youyun LIAN ; Fan FENG ; Xiang LIU

An active helium cooled structure was adopted in the high performance divertors, which requires metallurgical bond of plasma facing materials (PFMs) and structural materials. The tungsten materials were one important part in the PFMs, and the structural materials were made of reduced active ferritic–martensitic steels (RAFMS). The tungsten materials were joined to RAFMS using Fe-based amorphous reduced active fillers by vacuum brazing. Joint interface was analyzed by several methods, such as optical microscope, SEM, EDS. The results showed that The weld zone forms a good metallurgical bond with no voids or cracks. And then the mechanical properties testing were performed preliminarily. The shear strength is 250MPa, which is much higher than other results.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 58 / 23

Integral Benchmark Experiments on a Large Copper Block using GELINA accelerator to validate natural Cu neutron inelas-

neutron inelastic scattering cross sections from different neutron cross section databases.

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A neutronics integral benchmark experiment on a pure Copper block (dimensions 60x60x60 cm³), aimed at testing and validating recent nuclear data libraries has been performed at GELINA. GELINA is a powerful photo-neutron source using 75 kA, 110 MeV electron current impinging on a depleted ²³⁸U rotating target, producing a white spectrum with neutron energies ranging from epithermal region up to about 20 MeV with a mean energy of about 1.4 MeV and intensity up to 3.2E13 n/s. A large natCu block has been positioned at 100 cm from the target. The block had seven positions at different depths respect to the main neutron propagation direction where thin activation foils were used as neutron flux probes. Materials which are activated by different neutron energies were used and the measured fluxes were compared with calculations performed with MCNP5 neutron transport code employing different neutron cross sections database for comparison. With the MCNP5 it was modelled the neutron spectrum produced by GELINA accelerator and the neutron transport inside the block describing all the most relevant components of the experiment. This is the first time that a neutronics integral experiment on Copper is performed using such a white neutron spectrum and the results of the our comparison are used to validate the neutron inelastic scattering cross sections.

Eligible for student paper award?:

No

T.OA3: Blankets and Tritium Breeding: Liquid Breeders / 24

THE APPLICATION OF NANO FLUID TECHNOLOGY ON MHD EFFECT OF LIQUID METAL TRITIUM BREEDER BLANKETS

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The breeding blanket is the key nuclear component for power extraction, tritium fuel sufficiency and radiation shielding in fusion reactors. Using pure lithium (Li) or Li-containing liquid metal (e.g. eutectic alloy lead-lithium, PbLi) in fusion blankets as breeder is a very attractive option due to their high heat removal, adequate tritium breeding ratio, relative simple design, potential attractiveness of economy and safety. All liquid-metal blankets have special features associated with the nature of liquid breeders, including their high chemical reactivity, and especially interaction with the plasma-confining magnetic field. Flowing liquid breeder under magnetic field would result in various magnetohydrodynamic (MHD) phenomenon such as huge MHD pressure drop, quasi-two dimensional turbulence. It would be an effective way to reduce MHD effects by reducing electric

conductivity of liquid metal breeder.

A potential technology to reduce electric conductivity is nano fluid technology, which adds functionalized nanoparticles into fluid to change its physical property. We demonstrated that it is possible to reduce electric conductivity of liquid metal by adding electrically insulating nanoparticles. The liquid metal we tested was eutectic alloy of GaInSn, which is liquid at room temperature. The nanoparticle we chose was SiO₂, whose electric conductivity was several orders of magnitude lower than that of liquid metal and had a good wetting property with GaInSn. SiO₂ nanoparticles smaller than 200 nm in diameter were added into liquid metal GaInSn, forming dilute suspensions called nanofluids, which aimed to reduce the electric conductivity of the liquid metal. The nanoparticle weight fraction dependences of electric conductivity for GaInSn with fractions 0.05%, 0.1%, 0.2%, 0.5%, 1% were investigated and the electric conductivity measured from electrochemical workstation monotonically decreased with increasing nanoparticle fraction. The nanoparticle scale dependences of electric conductivity for GaInSn with particle nanometer scales 10nm, 20nm, 50nm, 100nm, 200nm were investigated and showed a weak relation. The best result we got was the case of 10nm with 0.5% weight fraction, where the electric conductivity was reduced by 4.25 times.

Based on this study, we evaluated the MHD pressure drop of typical liquid metal blankets such as DCLL and DFLL blanket, and the pressured drop was significantly reduced, which means that nano fluid technology was fit for liquid metal blankets. Further planning was scheduled for tritium breeder PbLi and Li.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 1 / 26

SAFETY ANALYSIS OF HELIUM COOLED CERAMIC BREEDER TEST BLANKET SYSTEM

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Institute of Nuclear Energy Safety Technology (INEST), Chinese Academy of Sciences (CAS) is one of the leading teams undertaking its corresponding research and development and mainly responsible for structure material development and safety analysis. After the kick-off meeting for PD phase of Helium Cooled Ceramic Breeder (HCCB) Test Blanket System (TBS) held by CN DA in early 2016, safety analysis becomes more and more important.

As an important part of the HCCB TBS safety assessment, accident analysis will be presented in this paper with the updated identification of reference accidents based on the approved version of preliminary safety report by IO, and more scenarios will be simulated and then analyzed using the thermal hydraulics code RELAP 5 and MELCOR. The results comparison of RELAP 5 and MELCOR will be done. The code-to-code comparisons can help identify code issues or implication errors that could go unrecognized. To understand the expected impact of modeling and data uncertainties, the uncertainty analysis approach of RELAP5 is Best Estimate Plus Uncertainty (BEPU) and will be extended for HCCB TBS to provide a direct understanding of the contribution of variations to specific parameters. The inventory of tritium will also be calculated under normal operation and its release under maintenance. To estimate consequences of airborne radioactive releases after accidents, the atmospheric radioactive transport and related potential for exposure will be calculated by MACCS combined with MELCOR. The primary objective of the above analysis is to evaluate the consequential radiological doses outside the ITER facility in scenarios selected to envelope all conceivable events, and thereby demonstrate compliance with the General Safety Objectives of the project.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 38 / 27

ITER PF6 Dummy Double Pancake Winding

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The Poloidal Field (PF) coils are one of the main sub-systems of the ITER magnets. The PF6 coil is being manufactured by the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP) as per the Poloidal Field coils cooperation agreement signed between ASIPP and Fusion for Energy (F4E). The ITER PF6 winding pack is composed by stacking of 9 double pancakes. Dummy double pancake fabrication will be implemented to qualify the critical fabrication processes, before series production. This paper describes the ITER PF6 dummy double pancake winding with a "two-in hand" configuration. Conductor de-spooling, straightening, ultrasonic cleaning, sandblasting, bending, turn insulation wrapping and deposition were carried out through the whole winding process. Joggles forming with high accuracy, helium inlets manufacturing with automatic welding and x-ray test, tail manufacturing with on-line assembly and PAUT test were performed. High synchronization of each unit in one line and between two lines were achieved. Tight tolerances of turn positioning and deviation between turns were obtained. The dummy double pancake winding has been completed with radial build-up deviation of 1mm, which met the requirement of less than 3mm.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 92 / 28

Analysis of Short Circuit Fault for 4.6GHz/6MW LHCD High Voltage Power Supply

Author: Yang Zhigang¹**Co-authors:** Fu Peng¹; Guo Fei¹; Sun Haozhang¹; Zhang Jian¹; Wang Mao¹; Liu Fukun¹; Huang Yiyun¹¹ Institute of Plasma Physics, Chinese Academy of Sciences**Corresponding Authors:** yangzhigang@ipp.ac.cn, fupeng@ipp.ac.cn

4.6GHz/6MW Lower Hybrid Current Drive (LHCD) is one of plasma current heating methods for Experimental Advanced Superconducting Tokamak (EAST). High Voltage Power Supply (HVPS) is the power supply subsystem of 4.6GHz/6MW LHCD system, which was designed, built and accepted successfully by Chinese National Development and Reform Commission in 2015. Then the new system has been in use for the 2015 EAST campaign. This paper presents the structure of 4.6GHz/6MW LHCD-HVPS and its transient operation state when its klystron load has short circuit fault. In order to protect the klystron and HVPS itself, the short-circuit fault and its transient process are analyzed and calculated in detail. And a three-electrode gas switch has been built to eliminate the short-circuit

fault in microseconds. In addition, the effectiveness of the three-electrode gas switch has been verified by simulation and experiment result. The HVPS has been used in 4.6GHz/6MW LHCD system and it has good performance for the entire 2015 EAST campaign.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 93 / 31

Design of Inverter Module on RMP coil Power Supply in EAST

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In Experimental Advanced Superconducting Tokamak (EAST), Resonant Magnetic Perturbation (RMP) coils, which are powered by RMP coil Power Supply (PS), are set to research Edge Localized Mode (ELM), Resistive Wall Mode (RWM) and Error Field Correction (EFC). The RMP coils are grouped in 8 sets with 2 coils in series as 1 set. And the 8 sets of coils are respectively powered by 8 sets of PSs. To generate a wide range of perturbation, the RMP coils will be supplied independently by DC or AC with the amplitude up to 2.5 kA in the frequency range of 50 Hz to 1 kHz. The maximum output voltage is 408 V. For further research of physicists, the output voltage and current should be up to 450 V and 4 kA. The current latency time should be less than 0.35ms (0-2.5 kA50%) *while the voltage latency time should be less than 0.25ms (0-450 V50%)*. And the rise rate for voltage and current should be larger than 300 V/ms and 5 kA/ms. The voltage ripple should be less than 2% of output voltage, when the output voltage is less than 5% of the rated value, the voltage ripple should be less than 10%. To meet the design requirements of output current, output voltage, current and voltage latency time and output voltage ripple, the switch frequency of IGBT, equivalent output frequency, RC compensation branch and topology structure should be designed carefully. The specific design progress including theoretical analysis and simulation results are given. The test results about different design requirements validate the correctness and reasonability of the scheme.

By applying carrier phase shifting PWM between branches, the ripple of output voltage can be reduced, and the equivalent switch frequency has increased. And the whole 8 sets of RMP coil PSs have been put into use in the latest EAST experiment. Until now, they can have an effect on ELM control according to the physical experimenters.

Eligible for student paper award?:

Yes

M.OA1: Experimental Devices I / 32

Development and Application of High Intensity D-T Fusion Neutron Generator HINEG

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Fusion energy becomes essential to solve the energy problem with the increase of energy demands. Although the recent studies of fusion energy have demonstrated the feasibility of fusion power, it commonly realizes that more hard work is needed on neutronics and safety before real application of fusion energy. A high intensity D-T fusion neutron generator is keenly needed for the research and development of fusion technology. However the intensity of D-T neutron generators currently on operation around the world is lower than 10^{13} n/s, which is severely restricting the research capability.

The Institute of Nuclear Energy Safety Technology (INEST), Chinese Academy of Sciences (CAS) has launched the high intensity D-T fusion neutron generator (HINEG) project to develop an accelerator-based D-T fusion neutron generator with the neutron yield higher than 10^{15} - 10^{16} n/s. HINEG consists of two phases: The first phase, named HINEG-I, aims to have the intensity of 10^{12} - 10^{13} n/s in order of magnitude, and the second phase, named HINEG-II, is designed to reach a neutron yield of 10^{15} - 10^{16} n/s via high-power tritium target system and high-intensity ion source. HINEG-I has been completed and commissioning with the neutron yield of up to 10^{12} n/s, while the related research on the key technologies of HINEG-II are on-going. HINEG can be used for research and development of nuclear technology and safety, including the validation of neutronics method and software, radiation protection, materials activation and irradiation damage as well as neutronics performance of components. Its application can also be extended to nuclear medicine, radiotherapy, neutron radiography, and other nuclear technology applications. This contribution will summarize all the latest progresses and future plans for the research and development of HINEG.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 94 / 33

Designing a Power Module for Compressed Plasma

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Abstract—Magnetic compression (MC) technology was suggested for tokamak to study compressed plasma in [Li, G., Scientific reports, 2015, 5] and a power module is designed here for developing its server power supplier. The minor-radius compression is one of the most effective method to improve the performance parameters of existing tokamaks, enabling the plasmas operated at high density, high temperature and high beta. In this paper, a high frequency and high power AC/Pulse converter is proposed, used for powering coils of minor-radius magnetic compression within vacuum chamber of the experimental advanced superconducting tokamak (EAST). The basic of the power module is a AC/Pulse converter of buck type, implemented by full-bridge phase-shift circuit and controlled by pulse-width-modulated (PWM). Also, control method adopts current closed loop Proportional-Integral (PI) control, has less than 1 ms current response time in real time. The converter is analyzed and the design procedure is discussed. Experimental results obtained from a 3kA converter prototype are presented to validate the converter's performance with the re-designed control board.

Keywords—magnetic compression; phase-shift PWM; AC/Pulse converters; tokamaks; fusion plasma; Lawson trinity parameter.

References

1. Li, G. High-Gain High-Field Fusion Plasma. Sci. Rep. 5, 15790; doi: 10.1038/srep15790 (2015).

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 35 / 34**Thermal and mechanical analysis of the Wendelstein7-X cryo-vacuum pump plug-in****Author:** Zhongwei Wang^{None}**Co-authors:** Gunnar Ehrke ; Boris Mendelevitch ; Jean Boscary ; Reinhold Stadler ; W7-X Team**Corresponding Authors:** zhw@ipp.mpg.de, reinhold.stadler@ipp.mpg.de, gunnar.ehrke@ipp.mpg.de, jean.boscary@ipp.mpg.de, boris.mendelevitch@ipp.mpg.de

The function of the cryo-vacuum pump (CVP) system is basically the control of the plasma density by condensing undesirable gases together with a set of turbo molecular pumps. One CVP will be installed under each of the 10 units of the actively cooled divertor in Wendelstein7-X for the long pulse operation up to 30 minute duration scheduled in 2020. The 10 CVPs are independent and each one is operated with supercritical Helium (ScHe) at 3.5K and liquid nitrogen at 77K fed by a plug-in, which is installed inside a dedicated W7-X port of the plasma chamber. The plug-in made of stainless steel provides for the vacuum boundary between the plasma chamber and the torus hall atmosphere. The outer dimensions of the plug-in are: ~ 2 m long and ~ 90 mm. 4 pipes (12 x 1 mm) are positioned inside the plug-in: 2 for the inlet/outlet of ScHe and 2 for the inlet/outlet of nitrogen, respectively. On the supply interface side, the pipes are equipped with bellows to compensate the thermal elongation during operation. The connection to the CVP is equipped with flexible hoses to allow compensating of assembly tolerances and to accommodate the displacement of the plasma chamber during operation. The design needs to guarantee the feeding at the specified temperature of ScHe and nitrogen while minimizing thermal losses and thermal interactions between pipes. Inside the plug-in the vacuum level is 10^{-3} Pa at RT and 10^{-5} Pa during operation. The pipes of the ScHe are shielded with a multi-layer super-insulation. In addition the cryogenic feed lines are protected with a cryo-shield against thermal loads in the port as well as in the plasma vessel. During baking, the relative displacement due to thermal expansion and mechanical load between the port and the cryostat could damage the plug-in and endanger the CVP feeding. This paper presents the thermal and mechanical analysis performed with ANSYS to check the selected design of the plug-in of the CVP.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 95 / 35**Implementation of an Excitation Controller for an Impulse Motor-Generator****Author:** Chi Wang¹**Co-authors:** Huajun Li¹; Mingnan Bu¹¹ Southwestern Institute of Physics

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An energy-stored impulse motor-generator (MG) is used in power supply system of HL-2A tokamak to produce short high-voltage or high-current surges of desired parameters that are usually used for magnetic field coils and auxiliary heating equipment loads. The operation changing of these loads will cause disturbances in generator's terminal voltage and the remark drop in MG's rotating speed. This paper describes the implementation of an excitation controller using LabVIEW and CompactRIO for the 125MVA impulse MG in power supply system of HL-2A/2M. Both a staged control strategy and digital PID algorithm built in LabVIEW are applied to the excitation controller that runs preciously in a one millisecond cycle to achieve voltage feedback control. The proposed excitation controller, composed mainly of host computer and CompactRIO embedded reconfigurable system, is available to restore and stabilize terminal voltage in accordance with the desired voltage waveform set by operators when pulsed loads and motor speed change quickly, also implements real-time monitoring of the working condition and some electrical parameters of excitation system and communicates with central control system via Ethernet to either download discharge control files at time interval between two impulse discharges or upload waveform data generated in control process after a pulsed discharge. Engineering experiment results show that the use of excitation controller improves the voltage stiffness of the power supply system and provides effective control of generator's terminal voltage under the management of central control system.

Eligible for student paper award?:

Yes

R.OP6: Safety, Operations, and Maintenance / 36

Fusion R&D Activities at INEST

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China has long been actively performing the integrated studies not only on R&D for ITER, but on the future fusion reactor. Institute of Nuclear Energy Safety Technology (INEST), Chinese Academy of Sciences (CAS) concentrates on the nuclear technology and safety, especially related to innovative fusion reactor/blanket concepts, neutronics simulation and experiments, materials and blanket technology, fusion safety assessment and regulatory etc. In this paper, recent research progress on fusion nuclear technology and safety at INEST will be presented.

The Super Monte Carlo Program for Nuclear and Radiation Simulation (SuperMC) has been independently developed at INEST and has been widely used in more than 30 major international nuclear projects. The development of High Intensity D-T Fusion Neutron Generator (HINEG) has been launched by INEST. The first phase (HINEG-I) has been completed and commissioning with the intensity of the order of 10^{12} n/s, while the key components R&D of the second phase (HINEG-II) with the intensity of 10^{15} - 10^{16} n/s is ongoing. China Low Activation Martensitic (CLAM) steel reached 6.4 tons industrial scale production, and is considered as the primary candidate for CN ITER-TBM. Meanwhile, joining and assembly techniques for CLAM steel were developed and a 1/3 scaled DFLL-TBM mockup was successfully fabricated. Dual coolant thermal hydraulic integrated experimental loop (DRAGON-V) is under construction to support the engineering design validation of PbLi breeder blanket with its parameters covering the design requirements of ITER-TBM. Moreover, INEST combining the domestic and international efforts to establish safety assessment methodology and safe design guideline for fusion energy development. In the recent research, the main scientific and technological safety gaps between the on-going ITER project and fusion demonstration reactor (DEMO) have been identified and the corresponding implications for the design and operation of DEMO are discussed.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 39 / 37

Electronic Transport Properties of NbTi in Cooper Matrix Superconducting Wires for ITER Applications

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The International Thermonuclear Experimental Reactor(ITER) device should demonstrate the scientific and technological possibility of commercial fusion energy production in large scale in order to solve the worldwide energy problem in the future. The superconducting magnet system is the key part of the ITER device to supply high magnetic fields for confining the deuterium–tritium plasma. The multifilament NbTi wires extruded in a Cu matrix with high quality have been studied to meet the specifications of superconducting strands. This work is presenting the study of signal-to-noise assessment, the electronic transport properties of NbTi wires extruded in a Cu matrix with 0.4mm in diameter and volume ratios of NbTi:Cu = 1.35:1. Normal-state magnetoresistance, I-V characteristics and superconducting state critical currents are thoroughly investigated.

Additionally, the critical current density has been investigated as a function of temperature and field using the expressions for the critical temperature, critical magnetic field and pinning force in NbTi. The measurements undertaken in this research cover a range of the magnetic field between 0T to 7T at temperatures ranging from 1.9K to 10K.

In order to measure the electrical resistance down to cryogenic temperature (2K) a Physical Property Measurement System (PPMS) is being used. The measurements have been done in various magnetic fields, up to 7T. The values of the measured resistance are the bases of the calculating the electrical resistivity, critical current density (Jc) and pinning force (Fp).

Key words: superconducting wires, vortex matter, critical current density, pinning force

Eligible for student paper award?:

No

T.OP1: Power Supply Systems / 39

Fuzzy Controller using Circulating mode for ITER Poloidal Field (PF) AC/DC Converter System

Authors: Mahmood ul Hassan¹; Fu Peng¹; Song Zhiquqn¹; Humayun Muhammad²; Xiaojiao Chen¹; Zhang Xi-uqing¹

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This paper presents new technique using fuzzy logic controller to improve the performance of the ITER poloidal field (PF) converter systems. A fuzzy controller is considered for ITER PF converter system, using the conventional PI controller and Fuzzy controller (FC). The dynamic behavior and transient response of the PF converter system in circulating mode is proposed under normal operation by analysis and simulation. The analysis results determine that the fuzzy logic control can achieve better operation performance.

Eligible for student paper award?:

Yes

M.OP3: Next Step Devices, DEMO, Power Plants / 40**Conceptual development of K-DEMO, highlighting maintenance and support details of in-vessel components****Author:** Thomas Brown¹**Co-authors:** Peter Titus¹; Charles Kessel¹; David Mikkelsen¹; Hutch Neilson¹; Kihak Im²; Keeman Kim²¹ Princeton Plasma Physics Laboratory² National Fusion Research Institute, Republic of Korea**Corresponding Authors:** khim@nfri.re.kr, hneilson@pppl.gov, kkeeman@nfri.re.kr, tbrown@pppl.gov, ptitus@pppl.gov, dmikkelsen@pppl.gov, ckessel@pppl.gov

The Korean fusion demonstration reactor (K-DEMO) has progressed through early concept definition activities to establish machine parameters, an operating point and the definition of the major core components. A key part of the conceptual development activities centered on the in-vessel components and the concept definition of the blanket/shield system, its segmentation and support arrangement. These systems have a major influence in defining the overall K-DEMO configuration and planned maintenance scheme. Earlier concept details of in-vessel systems has been updated with the addition of planned heating and current drive details, added blanket penetrations and the addition of some of the external heating systems located outside of the device core. Further definition of the blankets and support systems also led us to revisit the structural analysis of the in-vessel system design performed earlier [1, 2]. With in-vessel systems further developed, an initial assessment of a remote maintenance approach to remove all in-vessel components through the vertical ports also was made. The results of this activity along with an overview of the latest K-DEMO general arrangement will be presented.

References

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 [2] P. Titus, et.al, "Structural Assessments of the K-DEMO Blanket Modules", 29th Symposium of Fusion Technology, September 2016, Prague, Czech Republic

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 96 / 41**Conceptual Design of a Bidirectional Hybrid DC Circuit Breaker for Quench Protection of CFETR****Author:** Shusheng Wang^{None}**Co-authors:** Zhiquan Song ; peng Fu ; Hua Li ; Yingyao Lu**Corresponding Authors:** yylu@ipp.ac.cn, lihua@ipp.ac.cn, fupeng@ipp.ac.cn, wangss@ipp.ac.cn, zhquansong@ipp.ac.cn

The conceptual design of quench protection circuit for CFETR (China Fusion Engineering Testing Reactor) rated for current up to 70 kA and voltage of 15 kV is presented. The proposed scheme is based on a mechanical switch paralleled to the controlled static breaker. Static breaker is composed of a IGBTs unit and four diode units in a rectifier bridge allows it to be used in both current directions. The feasibility of bidirectional static breaker such as the reliable turn-on IGBTs unit, voltage and current sharing of each IGBTs and effect of rectifier diode recovery characteristic are also investigated for conceptual design of quench protection. The voltage and current unbalance are discussed in detail by simulation analysis which includes the influence of the gate signal delay and stray inductance in each IGBTs branch. Finally, a discussion of the conceptual design of quench protection circuit is given.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 107 / 42

Design and Analysis of the High Power DC Water-Cooled Busbar Connecting Type

Author: Zhongma Wang¹

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High power aluminum water-cooling DC busbar is used to connect the converter and the reactor for the ITER PF converter. The water temperature difference between import and export is less than 20°C, and the temperature rise of the busbar does not exceed 70°C according to the ITER standards. Three DC busbar connection types is designed to resolve soft connection overheat under the rated current. This paper mainly presents the temperature rise of DC busbar connection in different water flow and different steady state current. A temperature simulation of directly weld connection, lamination weld connection and soft connection are investigated by finite element method (FEM). The test is carried out to verify the correctness of simulation. The comparison results of three type's shows that directly weld connection method has a better performance.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 97 / 43

The Design on Pulse Distributor and Its On-line Status Diagnosis for ITER PF Power Supply

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The International Thermonuclear Experimental Reactor (ITER) Poloidal Field (PF) power supply is consisted by thyristor-based phase-controlled converters to supply megawatt power for six PF coils. Each bridge arm of ITER PF converter is paralleled with 12 thyristors to withstand 27.5 kA rated current. For avoiding the electromagnetic interference, the trigger signals are transmitted by optical fiber from the pulse generator. The pulse distributor is designed in this paper to transfer the optical signal to high current electrical trigger pulses with less than 2 μ s rising time, thereby triggering the paralleled thyristors reliably and isolating 20 kV voltage from converter to controller. In addition, the pulse distributor on-line status diagnosis is also the key function. The signals of crucial on-line status will be encoded and transmitted by optical fiber to controller thus guaranteeing the safe operation of ITER PF converter system. The pulse distributor has been experimented in ITER PF ac/dc converter for two years and all functions have been effectively verified.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 98 / 44

Design a Suitable Test Scheme for Triggering Bypass Protection Test of ITER PF Converter Unit

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The external bypass, as an important components of the international thermonuclear experimental reactor (ITER) poloidal field converter unit (PFCU), will provide a freewheeling loop for the load current to protect the magnets and PF converter modules from being damaged by over-current and over-voltage under fault conditions. The triggering bypass protection test is used to verify that the designed bypass can be triggered normally and endure the rated load current. In this paper, a suitable test scheme is designed for triggering bypass protection test of ITER PFCU based on the PF converter integrated test platform in ASIPP. This test scheme includes the triggering bypass method, calculating the trigger angle of the converter in inverter and a method of reducing the bypass current to zero in the absence of ITER mechanical switch. The feasibility of the test scheme is successfully verified by the simulation results and test results, both of which prove that the external bypass of ITER PFCU can be triggered normally and withstands the load current for 100ms. Due to the integrated inductance of the actual DC output circuit, the actual transfer time of bypass current in the test is more than the simulation results.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 99 / 45

Transient stability analysis of a flexible generator used in fusion power plant

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The China fusion engineering test reactor (CFETR) could output 50~200 MW fusion power for demonstrating power generation. The overall structure of fusion power plant in concept design is similar to a pressurized-water reactor (PWR), except a double-fed induction generator with flywheel (DFIGF) has been proposed to replace the traditional synchronous generator used in a PWR. The heat source of fusion station is not constant in long pulse operation mode of Tokamak. Hence it will lead to frequent inrush in synchronous generator and power grid. Although a high power electric heating steam pressure regulator can mitigate the inrush, it consumes a lot of energy. The proposed DFIGF has good power regulation ability and large energy storage capacity, which suits the application when instability of heat source exists. The simulation model has been established and dynamic response analysis has demonstrated that DFIGF can ride-through the burning interval smoothly and provide a flexible connection with power grid.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 64 / 46

Physics and engineering progress of CFETR integration design

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Chinese Fusion Engineering Test Reactor (CFETR) aims to bridge the gaps between the fusion experimental reactor (ITER) and the demonstration reactor (DEMO). CFETR will be designed and operated in two phases. Phase I focuses on a modest fusion power of up to 200 MW, where steady-state operation and self-sufficiency will be the two key issues. Phase II aims for DEMO validation with a fusion power over 1 GW. A new design has been made by choosing a larger machine with $R = 6.6\text{m}$, $a=1.8\text{m}$, $BT= 6\text{-}7\text{T}$ recently. Over 1GW fusion power can be achieved and technically it is easier to transfer from Phase I to Phase II with the new design. Physics and engineering progress of CFETR integration design are introduced in this paper.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 100 / 47

Servo-Power-Controller Design Based on EPICS

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Abstract—Based on the Experimental Physics and Industrial Control System (EPICS), a servo power controller is designed to be inserted into the existing power supply for testing its function of measurement and control in real time. The power supply controller uses double closed-loop feedback control, in which the outer-loop feedback control uses dead time modulation (DTM) technology to adjust the output current by tracking the external control signal. In this paper, the principles of the power supply and its controller are introduced, and the DTM method is studied and verified using MATLAB simulation and experiments to develop control technology for a tokamak power supplier.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 101 / 48

The Development of a monitoring system for poloidal Field Power Supply

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Inspired by the ITER Control, Data Access and Communication (CODAC), Experimental Physics and Industrial Control System (EPICS) has been chosen for the control system. The monitoring system is a important subsystem of ITER poloidal Field(PF) Power Supply(PS) system which can real-time monitor and control PF power supply running , timely warning, abnormal positioning, support multi-user configuration operation and has a good cross-platform portability and scalability. All nodes operate in a synchronized manner by using Time Synchronization Network (TSN) which based on IEEE-1588 protocol, are connected via a dedicated timing network. And moreover, the monitoring system also acts the role of Plant System Host (PSH), which helps non-EPICS controllers to keep working in ITER PFPS control system. EtherCAT Fieldbus is used in the field layer, which improves the real-time performance and reliability of the system. The asynDriver is used to develop interfacing device specific code to hardware driver. The test shows that the monitoring system has good performance during experiments and convenient human-machine interface to satisfy the requirements of all the experiments. In this paper, a description is given of the prototype ITER PFPS remote monitoring system that has been implemented on the experiments.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 102 / 49

Operation Analysis of Impulse Current Mode on ITER High Power DC Test Platform with SVC System

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The ITER Poloidal Field (PF) converter is comprised of four converter bridges, DC reactors, DC switches, etc. and most are non-standard. To evaluate the performance of these equipments, an ITER high power DC test platform has been built to carry out the rated current test and impulse current test. The latter is significant to verify the fault suppression capability of product. The DC test platform can output the rated 400 kA impulse current. The principle design and structure of the DC test platform is introduced in this paper. In addition, the impulse current test procedure is also discussed. The transient large current in impulse current test can produce huge impact reactive power, which impacts the fundamental reactive power and power grid voltage drop. The effect is analyzed by theoretical calculation and simulation. In order to suppress the transient effects of reactive power, a Static Var Compensator (SVC) system is added, which consists of Thyristor Controlled Reactors (TCR) and Fixed Capacitors (FC) with rated compensation capacity 83.2 Mvar. An impulse experiment is implemented on test platform with SVC system. The results of theoretical calculation, simulation and experiment are compared, which demonstrate that the SVC system is effective in compensating impact reactive power and it also performs well on inhibiting the power grid voltage drop.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 103 / 50

Rectangular Magnetic Sensor Array for Current Measurement Based on Numerical Quadrature Method

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Due to the feasibility of analytic solution, abundant researches have been conducted to study the measurement of line current by magnetic sensor array with circular arrangement. Since the large size bus is hard to be simplified as line current, transducer with circular arrangement is no longer suitable for large current measurement, especially in power system for fusion magnet coils, which has a rated current of tens of kiloampere and an impulse current up to hundreds of kiloampere when short circuit appears. Based on numerical quadrature algorithms, a magnetic sensor array based current transducer with rectangular arrangement is proposed in this paper. For the sensor array, the dimension of the rectangle depends on the magnetic field distribution, while the quantity and arrangement method is deduced by numerical quadrature theory which can improve the measurement accuracy. In addition, the effect of installation error is also discussed in this paper. A transducer prototype is developed to measure the current, which is 420 kA, through bundle busbars with a total section size of 2990 mm×650 mm. Compared with high accuracy fiber transducer, the consistency of experimental results demonstrates the high accuracy and reliability of the proposed transducer.

Key words : Current measurement, magnetic sensor array, numerical quadrature theory, large size bus

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 36 / 51

Conceptual design of the torus cryopump for CFETR

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A torus exhaust pumping system comprising 6 identical cryosorption pumps for China Fusion Engineering Test Reactor (CFETR) is designed to provide specified pressure levels and throughputs for various plasma operation modes. A conceptual structural design is performed and recommendations are presented in accordance with the design requirements. Based on the analysis of conductance, the effective pumping speed is calculated and the influencing factors are discussed. The thermal loads of cryopanel and shields are then calculated to verify the feasibility of the concept and determine the consumption of coolants at steady-state operation. The temperature distribution of the cryopanel is displayed by a 3D thermal-hydraulic calculation and regarded as one of the boundary conditions to obtain the stress distribution of the cryopanel. Simulation results indicate that the temperature distribution and the maximum stress meet the design requirements proposed by the cryogenic properties of the gases pumped. Finally, the configuration and operation scheme of the torus cryopumps are given in accordance with the overall pumping characteristics. The conceptual design of the torus cryopump provides methods and experience for the overall design of CFETR in the future.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 48 / 52

A rapid non-destructive inspection method applied to EAST lower divertor by IR thermography technique

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Graphite is used as plasma facing material in EAST lower divertor consisting of hundreds of graphite tiles which are connected to heat sink using screw bolt currently. A soft graphite sheet is inserted between graphite tile and heat sink to improve the ability of thermal conductivity. To evaluate the quality of the thermal contact between graphite tile and heat sink, efficient non-destructive inspection is essential before assembling divertor to EAST device. This paper introduces a rapid non-destructive inspection method for EAST lower divertor by infrared (IR) thermography which records the surface temperature of each graphite tile. The poor quality of thermal contact can be examined by comparison of the transient thermal response of surface of graphite tiles in symmetric region of the same divertor module at a rapid switch from hot to cold water flowing in the tube welded to heat sink. Three-dimensional (3D) thermal finite element (FE) analyses have been performed to simulate the excellent quality of thermal contact and as a reference for the experimental observations obtained by IR thermography.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 27 / 53

The reliability design of CFETR divertor

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The design of divertor is one of the most challenges for CFETR as the max heat flux on the CFETR divertor would be more than 20MW/m² and the max plasma current is 1MA. Surface and structural damage to divertor due to the frequent loss of plasma confinement remains a serious problem for the tokamak reactor concept. The deposited plasma energy during major disruptions, edge-localized modes (ELMs), and vertical displacement events (VDEs) causes significant surface erosion, possible structural failure, and frequent plasma contamination. The duty cycle of CFETR should be at least 0.3~0.5. This suggests a very high reliability for divertor and other key subsystems. At the pre-conceptual phase the structure of divertor has to be improved by system and reliability analysis. The plasma facing components, supports and their joints have been optimized by results of reliability design. The reliability design uses probabilistic methods to describe the loads and operation conditions and estimates the frequencies of plasma events.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 91 / 54

The Protection Strategy Design and Implementation for ITER PF Converter System

Authors: Liansheng HUANG^{None}; Xiaojiao CHEN^{None}

Co-authors: Ge GAO ; Peng FU ; Shiyang HE ; Zejing WANG ; Jun SHEN ; Guanghong WANG ; Lili ZHU

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The International Thermonuclear Experimental Reactor (ITER) Poloidal Field (PF) power supply has 14 thyristor based ac/dc converter units to feed six PF coils. PF1 and PF6 coils are both fed by one four-quadrant converter unit respectively, while the PF2-PF5 coils are supplied power by three four-quadrant converter units connecting in series and under sequential control to reduce reactive power for each. The rated parameters for each converter unit is ± 55 kA and ± 1.05 kV. On account of the complex operation modes and the huge power of converter unit, any fault in converter might lead to escalation of fault and then damage the equipments in case of improper protective action or out of protection. Consequently, the protection strategy is definitely an indispensable and important part for ac/dc converter system. In this paper, the protection strategy is carefully designed based on the fault analysis including the current unbalance between two sharing current bridges, circulation current out of control, bridge and DC terminal over current, and other internal fault, etc. The ITER PF ac/dc converter has been manufactured and its control system prototype has been finished. All protection strategies have been experimented and effectively verified on ITER PF ac/dc converter in Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP).

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 28 / 55

RAMI analysis for PFCs of EAST divertor

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Experimental advanced superconducting tokamak (EAST) is a D-shaped full superconducting tokamak device. Plasma facing components (PFCs) of divertor are most likely to be damaged during the operation, which become failure easily and have to be replaced and repaired frequently. To be able to reach the design objective, an assessment of technical risks by means of RAMI (Reliability, Availability, Maintainability and Inspectability) of the PFCs has to be performed on the graphite tiles as examples.

A functional breakdown of the graphite tiles was performed in a bottom-up approach, which are described using the IDEFØ method. Reliability block diagrams (RBDs) were prepared to calculate the reliability and availability of each function under stipulated operating conditions. Failure Mode, Effects and Criticality Analysis (FMECA) of the graphite tiles was performed to evaluate potential causes of failures and their consequences. Criticality charts highlight the risks of the different failure modes with regard to the probability of their occurrence and impact on operations. It was assessed that the RAMI analysis results meet the EAST project requirement during this design phase and the result will be qualified further when the device is updated.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 40 / 56

NSTX-U IN-VESSEL CONTROL COILS DESIGN CONCEPT

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A successful conceptual design was completed to develop in-vessel control coils (a.k.a. Non-Axisymmetric Control Coils or NCC). The NCC coils are a series of saddle coils that are intended to satisfy a number of physics criterion including magnetic breaking, error field control, fast Resistive Wall Mode (RWM) control and ELM stabilization. Customized Mineral Insulated Cable (MIC) was selected for the conductor material. The MIC was made from oxygen-free copper conductor, high purity Magnesium Oxide powder insulation and 304 stainless steel sleeve cover. Sample MIC was purchased from an outside supplier and various tests conducted for design and performance verification including high voltage testing, which necessitated the development of special test terminations. A concept design was also developed for terminating the MIC ends using non-conductive and vacuum sealed technique. Two alternative designs were proposed for joints inside the vacuum vessel. The NCC Coils are designed to be mounted in-front of the primary passive plates and underneath the PFC tiles. The passive plates will be modified to accommodate the coils. A new PFC tiles design concept was developed using High-Z materials. New penetrations design on the vacuum vessel wall was developed to prepare one port per coil. A new power patch panel will be required to provide the ability for various combinations of connections between the NCC, the existing RWM Coils and the existing SPA power suppliers.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 108 / 57**New features of the W7-X Safety Control system for OP 1.2****Author:** Reinhard Vilbrandt¹**Co-authors:** Andreas Werner ¹; Jörg Schacht ¹; Hans-Stephan Bosch ¹; Georg Kühner ¹; Dirk Naujoks ¹; Sven Degenkolbe ¹¹ *Max-Planck-Institute for Plasma Physics, D-17491 Greifswald, Germany***Corresponding Authors:** reinhard.vilbrandt@ipp.mpg.de, sven.degenkolbe@ipp.mpg.de, joerg.schacht@ipp.mpg.de, andreas.werner@ipp.mpg.de, georg.kuehner@ipp.mpg.de, hans-stephan.bosch@ipp.mpg.de, dirk.naujoks@ipp.mpg.de

After the successful first operation campaign of Wendelstein 7-X in 2016 the experiment is being upgraded for the next stage called OP 1.2.

Due to some new or extended components like heating systems and diagnostics, the increased energy level in the machine, and some new safety requirements, the central safety system (cSS) has to be extended. In addition, a safety level based switchover is being implemented to support the engineer in charge and the cSS-operator in their assessment of enabling a predefined set of components in different operation modes of W7-X. Furthermore, the safety levels define the entrance conditions to the experiment areas. This safety level concept comprises only 5 different states, “off”, “stand by”, “experiment pause”, “experiment”, and “W7-X emergency stop”. For test and calibration purposes a certain set of components can be enabled in some special operational modes, like Laser or ICRH calibrations, boronization etc.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 37 / 58**The feasibility of application the existing IVVS concept to CFETR****Author:** ZIBO ZHOU¹¹ *CASIPP*

The Chinese Fusion Engineering Test Reactor (CFETR) is the next generation fusion device of China for realization of fusion energy. Its mission aims to bridge the gaps between the fusion experimental reactor ITER and the demonstration reactor (DEMO). To carry out the maintenance work of CFETR, remote handling systems shall be employed. To inspect the position and status of in-vessel components of CFETR, a remotely operated in-vessel viewing system (IVVS) shall be developed. A kind of IVVS prototype with an articulate arm and a viewing system has built in China, which bases on the size of EAST Tokamak. To test, install and store the IVVS around the Tokamak, a cylindrical vacuum vessel with foundation support inside is designed and checked. To connect the vacuum vessel to the Tokamak, an ultra-high vacuum gate valve a bellow is utilized between them. To keep the vacuum degree of the Tokamak after connection, a set of vacuum unit with molecular pump is configured for the vacuum vessel. Various performance tests of the IVVS are performed on the remote handling test facility before it connects with EAST Tokamak. The test result shows that it is feasible to apply this IVVS concept to CFETR.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 29 / 59

Upgraded Design of EAST Lower Divertor

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EAST is one of the most important experimental fusion devices of China, the design of each component has important reference significance for China Fusion Engineering Testing Reactor (CFETR). Divertor, as one of the most important in-vessel components on the EAST, has always been quite difficult and challenging for its design. With the completion of the upgrade of the upper divertor, EAST has achieved a series of good test results, and then, the thermal load capacity of the lower divertor has become a bottleneck that constrains EAST to obtain higher parameters. This article explains the upgrade of the lower divertor which referred to the structure of tungsten monoblock on upper divertor. The lower divertor uses the circular monoblock structure at hit point area, at the end of monoblocks, end boxes are applied. The role of the end boxes is rational distribution of water flow in the premise of maintain the existing water supply capacity (1.8KG/s), which will improve the thermal load capacity of lower divertor from $2\text{MW}/\text{m}^2$ to $5\text{MW}/\text{m}^2$. Finally, the structure of the target plate satisfies the requirements, at the same time, the existing space can also accommodate the installation of the support structure. The reconstruction of the lower divertor not only provides support for the subsequent physical experiments on the EAST, but also provides an important reference for the design of the CFETR divertor.

Key Words:EAST,lower divertor,monoblock,tungsten,thermal load capacity

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 30 / 60

Research and Analysis on the Compatible Structures of the CFETR Divertor Based on the Remote Handling Requirements

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Abstract: The Chinese Fusion Engineering Test Reactor (CFETR) is the next device in the roadmap for the realization of fusion energy in China, which aims to bridge the gaps between the fusion experimental reactor ITER and the demonstration reactor (DEMO). As a key in-vessel component in tokamak fusion reactor, the divertor which eliminates impurities, Helium ashes and heat flux often exposes to tritium environment and neutron radiation. Therefore, the divertor is very easily damaged, and it needs to be maintained regularly. Its maintenance should be handled by remote handling (RH) ways rather than by personnel directly, so the structures for remote handling ways should be suitable and feasible. It not only can meet the requirements for the RH but also should have adequate strength to resist high electromagnetic forces generated by Eddy current and Halo current due to the plasma disruptions. In this paper, the CFETR RH compatible structures are analyzed and two schemes are put forward based on the RH requirements. Then their working conditions, especially electromagnetic analysis are discussed to validate the design quality for the newly designed structures. After that, the maintenance processes of two schemes are simulated in the virtual environment by the software Delmia. Through simulation, the installation and dismantlement processes of the divertor can be vividly seen in Delmia where can check whether there will be interferences between the divertor and other components. What's more, the distribution of other components and the location of RH ports can be determined by simulation. And the maintenance procedures of the divertor are planned rationally according to the simulation PERT chart. Finally, the optimal path for the divertor RH process is chosen through simulation.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 92 / 61

A digital signal processing system of digital Rogowski current transducer with comb filter

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In ITER poloidal field (PF) prototype converter testing, the Rogowski current transducer is used to measure the current in a DC bus bar. When thyristors, which are parts of PF converter, are triggered on, they will produce electromagnetic noise around. The noise signal, which has a strong amplitude and fixed frequency, is easy to be coupled by cable between Rogowski coil and integrator, and transmitted to integrator. Then, the differential signal, which is produced by Rogowski coil and proportional to the current in DC bus bar, will be submerged. Consequently, it will lead to a very low signal-noise ratio, and the integrator cannot work. A digital signal processing system has been designed to solve the problem mentioned above. The design is based on the dual-ADC structure digital integrator which has been developed at ASIPP. A digital comb filter is utilized to filter out the electromagnetic noise signal, and measures are taken to weaken it from the hardware perspective. The experiment indicates that the method presented in this paper can decrease the amplitude of electromagnetic noise, increase the signal-noise ratio and improve the measurement accuracy.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 81 / 62

The Design of Real-time Communication System Based on RFM and MRG-Realtime for EAST

Author: Li Chunchun¹

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For the purpose of keeping a steady-state plasma and avoiding plasma instabilities, EAST plasma control system (PCS) needs more diagnostic data produced by the EAST distributed subsystems to perform complex control algorithms. To solve the transmission problem of the above data, a real-time communication system based on MRG-Realtime (Messaging, RealTime, and Grid) operating system and reflective memory (RFM) network is designed. It is expected to realize that the data acquisition (DAQ) systems of the EAST subsystems can transfer the diagnostic data to PCS in real time through RFM network during capturing data, then PCS uses the available data to carry out the necessary calculations according to control algorithms and sends the commands to the subsystems in every control cycle.

Then a small real-time communication system according to the prototype of the EAST vacuum DAQ system with sampling rate of 10 KHz was built for a performance test. In this test, the vacuum DAQ system transferred the acquired data to PCS in per control cycle of 100 us, which preliminarily meets the above mentioned requirements. The test was performed for 1000s to get a stable and reliable result, which described in detail in the paper. In addition, the future work is also discussed.

Eligible for student paper award?:

Yes

R.OP3: Tritium Extraction and Control / 63

Analyses Of DEMO Tritium Self-sufficiency

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DEMO tritium self-sufficiency will be one of the remaining challenges after ITER before achieving commercial fusion energy. Though ITER has not been set the tritium self-sufficiency target, tritium self-sufficiency related fusion science and technology are the maximum attainable level right now.

In this study, the dynamic tritium cycle was simulated both for steady and pulsed DEMO using the system dynamics platform to explain the dynamic tritium inventory and start-up inventory in different systems. Meanwhile, sensitivity analysis of tritium self-sufficiency was performed on the basis of ITER design parameters so as to explain the key factors and reachability of DEMO tritium self-sufficiency. After performing the evaluation, the key influencing factors of tritium self-sufficiency are not only blanket tritium breeding ratio (TBR), but also tritium burn-up fraction, tritium retention amount in the materials and detritiation efficiency of retired components materials. For a typical DEMO, a 50% burn availability and ITER related fusion physical and technological values (e.g. 1% of burn-up fraction, 1h of tritium cycle time, waste management strategy) can be predictive. Under the

condition, the required blanket tritium breeding ratio (TBR) is less than the achievable TBR designed in recent years.

Eligible for student paper award?:

Yes

W.OA3: Neutronics and Multiphysics Analysis / 64

Development and application of advanced nuclear software SuperMC for fusion

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SuperMC, developed by FDS Team in China, is a general, intelligent, accurate and precise program for the nuclear design and safety analysis. The latest version supports n, γ transport calculation, depletion calculation, activation calculation and shutdown dose rate (SDR) calculation, with the advanced features of automatic modeling, visualization and cloud computing.

Built-in activation calculation was newly developed based on a new matrix exponential method-Chebyshev rational approximation method (CRAM) method. Depth first searching based dynamic construction of activation chain was developed to avoid redundant nuclides. Adaptive reduction of massive matrix order was developed to accelerate coefficient matrix solving. The SDR calculation was developed based on both rigorous two step (R2S) method and direct one step (D1S) method.

To improve performance of transport calculation, several novel methods were developed. Global Weight Window Generator (GWWG) was proposed, in which the expected contribution to a uniform particle distribution was considered, and an automatic iteration scheme was implemented to speed up the weight window generation.

Hybrid geometry representation and modeling methods was implemented in SuperMC. Hybrid facet and constructive geometry modeling method has been implemented, which enables SuperMC directly using complicate CAD models including spline surfaces without pre-processing. Besides, unstructured mesh was newly applied in SuperMC to enhance the description capability of arbitrary shape and process the multi-physics coupling analysis. A new speedup method called feature size tree was presented to accelerate the unstructured mesh geometry processing and further reduce the memory consumption.

SuperMC has been verified and validated by more than 2000 benchmark models and experiments, such as SINBAD, ITER C-lite. Meanwhile, series of SDR benchmark tests were used to verify the activation and SDR calculation, such as the ITER shutdown dose rate benchmark and ITER-T426

shutdown dose rate experiment. The results of SuperMC agree well with other codes and experiment results. In addition, with the new developed acceleration methods, calculation speed of SuperMC was greatly increased by about 200 times for ITER Alite model.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 93 / 65

Development of Rotational Speed Control Equipment And Brake Equipment for 300MVA Pulse Generator

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For supplying enough power for 2M-HL Tokamak, a new 300MVA pulse generator has been developed, the new generator with 400 tons of rotor to stored energy will be driven by an 8500kW asynchronous motor . In order to reduce the large starting current, a high voltage variable frequency converter has been developed as the starting device because of the large inertia. Liquid resistors in series with motor rotor as the standby starting equipment has been developed. Two sets of equipment start the generator through the switch . In this paper, a simulation model of the high voltage variable frequency converter is built by the MATLAB/SMULINK. Calculation are made for motor rotor series of liquid resistors. The maximum series resistors , the starting current and starting time are obtained.

The working speed of 300MVA generator is 500RPM. It costs more than one hour to stop freely the unit. So a energy-consumed braking equipment and mechanical braking equipment are developed. These two equipments are analyzed in this paper. The braking resistor, excitation current and braking time are calculated. the mechanical brake pressure and brake time are calculated also.

During the debugging the unit, the actual running data and calculation data are compared. The analysis and calculation are more conform to the actual running situation.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 94 / 66

The Design of a 70kA/20kV Two-section Pyrobreaker for Quench Protection

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Due to the fast response, high reliability and one-time-use property, explosive driven circuit-breaker known as pyrobreaker(PB) has been applied in several high power supply systems. This paper presents the designing process of a new two-section PB with the capability of opening a DC current of 70 kA under a voltage up to 20kV. In accordance with the CFETR (China Fusion Engineering Test Reactor) specifications, this switch will be applied as a back-up breaker for quench protection.

Related simulations and calculations about the steady temperature rise, explosion process and commutation process are described in this paper, which fill the gap in the theoretical analyze for the designing of this kind of PBs. It will simplify the existing design method, which normally involves a great number of time-consuming and money-costing experiments. Several tests, including the steady state test and operation test, are conducted on the prototypes. The results verify the feasibility of the new model and demonstrate the reliability of the presented designing process.

Eligible for student paper award?:

No

M.OP1: Plasma Operation and Control / 67

Real-time detection and localization of magnetic island used for neoclassical tearing mode control and disruption mitigation

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It is well known that the neoclassical tearing modes (NTMs) can be destabilized by the perturbed bootstrap current, reducing the plasma confinement and even leading to a major disruption in both standard ELMy H-mode and advanced tokamak scenarios. In order to stabilize the NTMs with electron cyclotron resonant heating (ECRH) and electron cyclotron current drive (ECCD), the magnetic island is required to be localized accurately and then the EC beam power is deposited exactly inside the island. For the NTMs suppression on EAST tokamak, a real-time system to detect the magnetic island and trace its radial location has been developed. In this system, the diagnostic signals from electron cyclotron emission (ECE) and Mirnov coil measurement are acquired and processed in real time to obtain the frequency and amplitude of magnetic perturbation as well as the mode radial location; as an alternative, the soft-x ray signals are taken to deduce the mode location instead of the ECE diagnostic in the case that the low hybrid wave is applied to plasma heating and current drive. The construction and the algorithm implementation of the real-time system is introduced in this paper.

As the outputs of the real-time system, the mode radial location is provided to the ECRH launcher to determine the angle of EC beam injection. The island amplitude is used to control the gyrotron power on and off, and meanwhile involves in the feedback control of the ECCD deposition position with respect to the island position. Furthermore, the island amplitude is also monitored to generate a disruption alarm to activate the massive gas injection valve for the disruption mitigation, since in some cases the NTM suppression by the ECCD could fail due to an infaceable misalignment, the insufficient EC power for the mode stabilization and so on, and then the magnetic island would grow further leading to a major disruption. An integrated control strategy available to both NTM control and disruption mitigation is being developed and is expected to be presented.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 82 / 68

Modeling of Ohmic Disruptive Discharge in J-TEXT Using the Tokamak Simulation Code

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A simulation of J-TEXT Ohmic disruptive discharge has been achieved by using of the Tokamak Simulation Code (TSC) model. The anomalous transport of disruption was adjusted according to heat transport across large magnetic island. The simulation result of the plasma current, electron temperature, loop voltage and the disruption are compared with experimental disruptive discharge data. According to simulation result the dynamic response of one test plasma-facing module were calculated, the numerical results show that the maximum stress of the test module is in safety range.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 95 / 69

DESIGN OF CURRENT-PULSE POWER SUPPLY FOR TEARING MODE CONTROL ON THE J-TEXT TOKAMAK

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Tearing Mode (TM) creates magnetic islands in the tokamak. Using external resonant magnetic perturbation (RMP) coils is a convenient method to affect magnetic islands. To avoid mode locking and major disruption, the stabilization of TM control by using RMP is a promising method. A new method for applying modulated magnetic perturbation is presented to suppress magnetic island and accelerate island rotation. The phase difference between TM and external RMP is denoted by Φ . RMP has a stabilizing (destabilizing) effect on island when $0.5\pi < \phi < 1.5\pi$ ($-0.5\pi < \phi < 0.5\pi$) and an accelerating (decelerating) effect when $\pi < \phi < 2\pi$ ($0 < \phi < \pi$). Moreover, a net suppression effect has been proved by numerical simulation result when $\pi < \phi < 2\pi$. Based on this mechanism, if RMP is applied to the phase region of $\pi < \phi < 2\pi$, magnetic island can be suppressed and accelerated in every island rotation period.

J-TEXT tokamak has a set of RMP system which contains four sets of in-vessel saddle coils. To achieve the mechanism above, a bipolar current-pulse power supply with magnetic island phase detected system is applied to TM control. In the phase region of $\pi < \phi < 2\pi$, the power supply gives positive current-pulse to accelerate island, and in the region of $0 < \phi < \pi$, it gives negative current-pulse to double the effect. The island phase detection should be accurate and current-pulse power supply should have rapid current changing edges to have expected effect. In this paper, the working principle of the current-pulsed power supply is elaborated. The power supply contains a H-bridge inverter using IGBTs to provide high power high frequency bipolar current for inductive load. A six-pulse rectifier with a LC filter is used for DC source. A DC/DC chopper is added on bus to have a faster response of adjusting load current amplitude. Before the H-bridge, a set of boost capacitors with a diode is designed to steepen current changing edges. It will store the energy of inductive load on current falling edge and boost voltage on current rising edge. To ensure phase region accurate, current edge changing should be less than 100 us. The current frequency should follow the TM frequency changing from 1 kHz to 7 kHz and amplitude should be 3 kA in maximum. Based on calculation and simulation results, the capacitance should be suitable to keep the balance between current changing speed and capacitors voltage. A power supply prototype has been made to obtain experiment results. Because of the leakage inductance on bus, there is a voltage spike at the IGBT turn-off moment. A snubber circuit is designed for inverter to reduce the voltage spike.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 2 / 70**Experimental Study on Natural Circulation Heat Transfer of Square Channel in Water Cooled Blanket****Authors:** changhong peng¹; hui bao²; songlin liu²; yun guo¹¹ *Institute of Nuclear Science and Technology, University of Science and Technology of China*² *Institute of Plasma Physics, Chinese Academy of Sciences***Corresponding Authors:** baohui92@126.com, guoyun79@ustc.edu.cn, pengch@ustc.edu.cn, slliu@ipp.ac.cn

Square channel is widely used in the conceptual design of water cooled blanket of fusion reactor for cooling and providing appropriate inner temperature field for tritium breeding. Blanket is one of the most important safety components and thermal-hydraulic characteristics of blanket directly determine the heat transfer efficiency and safe operation of fusion reactor. Under accident conditions, the natural circulation phenomenon occurs without any mechanical devices intervention when the field forces acting on the fluid produce density gradients able to induce natural convection, which is the main heat transfer mechanism and a important measure to mitigate consequence of the reactor accident. For square channel(8mm*8mm) in blanket, the experimental study of natural circulation heat transfer has been conducted. Experimental results showed that natural circulation flow was not a independent parameter, which increased with the increase of the heat flux and the decrease of system pressure within the experiment scope. Simultaneously, natural circulation heat transfer was strongly affected by system pressure and heat flux. A new correlation was developed on the foundation of experimental data, which could predict the heat transfer coefficient of natural circulation with the maximum relative error of 30%. Comparing the experimental results with the results of forced circulation, it could be found that the heat transfer coefficient of natural circulation was lower than the heat transfer coefficient of forced circulation.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 41 / 71**Test upgrade for ITER HTS current leads series production****Authors:** Kaizhong Ding¹; Chenglian Liu¹**Co-authors:** Pierre Bauer ²; Kun Lu ³; Yuntao Song ¹; Qingqing Du ⁴; Tingzhi Zhou ¹¹ *Institute of Plasma Physics, CAS*² *ITER Organization*³ *Institute of Plasma Physics, CAS*⁴ *Institute of Plasma Physics***Corresponding Authors:** kzding@ipp.ac.cn, lukun@ipp.ac.cn, liucl@ipp.ac.cn, pierre.bauer@iter.org, songyt@ipp.ac.cn, duqing@ipp.ac.cn, tingszhou@ipp.ac.cn

Institute of Plasma physics, Chinese Academy of Siciens (ASIPP) is being in charge of the the manufacturing and cold testing of the 60 single HTS current lead series production for ITER device in 2017. The norminal current for these HTS current leads are from 10 kA to 70 kA. The cold testing

of all 3 types of ITER lead prototypes were qualified by ASIPP from 2015 to 2016. This paper will firstly summarize the testing items and testing result of ITER lead prototypes, then based on the prototypes experiences, the final test piping & instrument diagram for ITER lead series production is presented and discussed. A additional new test platform in 5K powering test facility for the series is discussed here to optimize the test arrangement. The 30 kV high voltage isolated instrument for the cryogenic temperature and voltage are firstly used to protect the CODAC test system. The detail test procedures and test items for the series will be discussed. At present, the manufacturing of the first pair of ITER lead will be started in the early of 2017. Author would like to present the test results based on the test upgrade work in the conference.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 65 / 72

Conceptual design of the cryogenic system for CFETR

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China is planning to construct the China Fusion Engineering Test Reactor (CFETR), in order to bridge the gap between ITER and the Prototype Fusion Power Plant. The cryogenic system of CFETR is indispensable to provide cooling for the superconducting magnet system and cold structures at 4.2 K, the cryopumps at 3.7 K, the HTS current lead at 50 K, and the thermal shields at 80 K. In this paper, the magnet system of CFETR is introduced and the cold mass is estimated. The cryogenic system heat load is extrapolated from that of ITER. The required refrigeration capacity of the cryogenic system is evaluated, taking into account the dimension of CFETR and especially the large burn duty cycle of 50%. The conceptual design of the CFETR cryogenic system is introduced.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 59 / 73

Neutronics and thermomechanical analysis of a conceptual shielding blanket for CFETR

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Co-authors: Lei Li ; Changle Liu ; Damao Yao ; Xiang Gao

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The conceptual design of Chinese Fusion Engineering Reactor (CFETR), which will be operated in two phases has been proposed to fill the gap between ITER and DEMO plant [1]. In order to maintain the performance of Vacuum vessel (VV) and external machine components especially the superconductors, a reasonable shielding blanket is needed between tritium breeding blanket (BB) and those superconductors. On account of different BBs, the shielding blanket may have different structures. So far, mainly three conceptual BBs have been designed for CFETR: Helium gas cooled T breeding blanket, water-cooled and LiPb liquid metal coolant blankets. In this paper, based on the water-cooled BB, a conceptual shielding blanket (SB) structure has been proposed. The SB module

structural materials are investigated, and the detailed shielding scheme is researched with the simplified 3 D model by using the MCNP program. The cooling structure of the SB module is provided, and the thermomechanical behaviors are analyzed by finite element method. These neutronics and thermomechanical analysis results indicate that the conceptual SB module meets the requirement for shielding and in compliance with thermomechanical standards of structure materials.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 3 / 74

Li₂Be₂O₃ pebbles prepared via sol-gel method: multifunctional blanket material designed to both tritium breeder and neutron multiplier

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As a notable multifunctional material in breeding blankets, Li₂Be₂O₃ could play a role on both tritium breeder (lithium) and neutron multiplier (beryllium), which is being considered for use in increasing the tritium breeding ratio. However, synthesis of Li₂Be₂O₃ ceramic powder still needs to be further studied, and the preparation of Li₂Be₂O₃ pebble-type breeder has not been reported in recent reports. In this paper, two-step method is carried out: firstly, Li₂Be₂O₃ ceramic powder is synthesized by sol-gel method. Stoichiometric CH₃COOLi and Be (NO₃)₂ are dissolved in distilled water and transfer to a gel by aqueous solution polymerization of acrylamide. Then the nano-sized ceramic powder is observed by sintering process at high temperature in the air. The sintering temperature is established by thermogravimetric/differential scanning calorimetry (TG–DSC). The Li/Be molar ratio of products are detected by inductively coupled plasma atomic emission spectroscopy (ICP-AES). Crystal structure of this powder is characterized by the X-ray diffraction spectroscopy (XRD). Secondly, ceramic injection molding methods in several conditions are used to prepare the Li₂Be₂O₃ pebbles. The ceramic slurry is prepared by milling 85 wt% synthesized Li₂Be₂O₃ powder and 15% polymerizable binder solution (acrylamide and N,N'-Methylenebisacrylamide). Then the slurry is doped to the liquid paraffin with the temperature at 80 °C, and each drop is about 10μL. The liquid drops could be solidified during the sedimentation in the hot oil. The obtained gel balls could transfer into ceramic pebbles after the second sintering and the strength of the pebbles is also measured by mechanical tester. The results show that when the concentration of polymerizable binder solution is selected about 10 wt%, the resulting pebbles had a better spherical shape and a higher strength.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 42 / 76

Performance test of CICC joint for ITER correction coil

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Abstract: In the frame of CICC testing for correction coils (CC) of ITER, the soldered joint design was developed and tested up to 12kA in a loop comprising the secondary winding of a superconducting transformer. The transformer which consists of two concentric layer-wound superconducting solenoids with the primary inside secondary coil was designed and manufactured. The primary coil was wound by 0.87mm diameter multifilamentary NbTi wires and secondary coil was wound by ITER CC conductor. The quench protection system was also introduced. The joint was test in liquid helium (LHe) temperature. The hall sensor was installed on the CC conductor to measure the current of secondary loop. Test results are present and showed that the joint resistance remained about 2nΩ in the current range from 8kA to 12kA, which was satisfied the requirement of ITER CC's design.
Index Terms: superconducting transformer, CICC, NbTi, joint resistance

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 85 / 78

Simulation study of large power handling in the divertor for CFETR phase II

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The Chinese Fusion Engineering Testing Reactor (CFETR) is the next device for the Chinese magnetic confinement fusion (MCF) program that aims to bridge the gaps between the fusion experiment ITER and the demonstration reactor DEMO. CFETR will be operated in two phases: Steady-state operation and tritium self-sustainment will be the two key issues for the first phase with a modest fusion power of up to 200 MW. The second phase aims for DEMO validation with a fusion power over 1 GW. For meeting both Phase I and Phase II targets and easily transitioning from Phase I to Phase II with the same machine, new design has been made by choosing a large machine with $R = 6.6\text{m}$, $a=1.8\text{m}$, $BT=6\text{-}7\text{T}$ since 2015. So far, most physics design are centered around the aims of phase I [1].

The ability to exhaust the plasma power loss is a critical issue to the successful production of a fusion power reactor. In fact, for phase I fusion power of $\sim 200\text{MW}$ is less than ITER, it would not be a serious challenge with a ITER-like W/Cu divertor. However, during Phase II of CFETR, exhausted thermal heat from the core plasma (P_{sep}) is expected to be larger than 200 MW, which is increased for the steady-state operation scenario since larger current drive power is injected into the core plasma. At the same time, the divertor heat load will extremely exceed the material tolerable limit ($\sim 10\text{MW}/\text{m}^2$), which could prevent the long pulse or steady state operation. Externally seeded impurities can help partially radiate the heat before it reaches the divertor. The seeded impurities however cannot be so large as to negatively impact the plasma performance in the core. We have simulated the baseline operation scenario parameters by using SOLPS5.0 (B2.5-EIRENE) code package for a standard lower single null (LSN) divertor configuration. The modeling shows that Ar (or Kr) puffing is highly effective in mitigation of the divertor peak heat flux. In addition, the radiation loss fraction inside the separatrix will enhances and leads a reduction of the power across the separatrix entering into scrape-off-layer region P_{sep} as impurities puff rate increase. The edge effective charge Z_{eff} and P_{sep} which are experimentally proved to be closely related with core confinement factor H_{98} [2,3] are studied with Ar and Kr seeding as well. Comparison simulations of different divertor geometries will be performed in this work to optimize CFETR design.

Further work about advanced divertor configuration (field expansion and increase in connection length) together with technical improvement of heat load removal capacity will be study in future.

Acknowledgements:

This work is supported by National Magnetic Confinement Fusion Science Program of China under contract no. 2014GB11003, 2015GB101003, 2015GB101000; National Nature Science Foundation of China under contract no. 11305206.

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Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 70 / 79

Progress in the development of CFC/CuCrZr components for HL-2M divertor

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HL-2M is a new medium-sized copper-conductor tokamak device under construction at Southwestern Institute of Physics and can perform advanced divertor configurations, such as snowflake and tripod. An open cassette divertor structure with active water cooling has been designed to meet the operation requirements of HL-2M tokamak. The divertor consists of a flat-tile CFC/CuCrZr component and a cassette structure. The CFC/CuCrZr component is made of a water-cooled CuCrZr copper alloy heat sink armored with CFC tiles CX-2002U. The CFC surface was modified by using slurry technique to improve its wettability to copper. An oxygen-free copper (OFC) buffer layer was cast on the modified CFC surface in order to mitigate the internal stresses caused by mismatch in the coefficient of thermal expansion of CFC and CuCrZr. Vacuum brazing of OFC/CFC tiles to CuCrZr heat sink was performed by using a silver free brazing alloy. Non-destructive examination followed by high-heat-flux testing was performed to access the manufacturing quality of the joint interfaces between the CFC tiles, OFC and the heat sink. The CFC/CuCrZr components experienced cyclic tests of 7-10 MW/m² for 1000 cycles without visible damages. High-quality bonding between CFC and the heat sink was achieved to ensure the heat removal capability of the components.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 96 / 80

An Integration method of Hybrid Power Filter for Specific Harmonic Suppression in Tokamak Power System

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This paper deals with an integration method of hybrid power filter in Tokamak Power System. The integration method not only takes full advantage of SVC which originally installed in Tokamak but also gives a combination of an active power filter to suppress the selective harmonics. Due to the rich harmonic spectrum in the AC side of non-linear load in Tokamak Power Systems, along with most reactive power compensation and harmonic filtering platforms in Tokamak barely notice low-order harmonics especially below the third harmonic, these low-order harmonics can be resonate with the capacitive impedance and inductive impedance in the circuit. The resonance will do harm to the grid system and the Tokamak system. Theoretical analyses and simulation results obtained from the EAST power system evaluate the effectiveness of the Integration method and stability of the whole hybrid power filter system. In addition, the simulation results are validated by experiments based on a testing platform.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 49 / 81

The forming die design and experimental research of CFETR Vacuum Vessel shells

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Abstract : Vacuum Vessel is the main component of CFETR (China Fusion Engineering Test Reactor), a series of theoretical studies and engineering verifications should be done from the period of Vacuum Vessel structure design to analysis optimization, from manufacture and assembly process to the plasma experiment study. Notably, shells forming is relatively in the earliest phase, also one of the most important processes during Vacuum Vessel manufacturing. The profile errors caused by spring-back of shells at room temperature will directly influence the subsequent manufacturing techniques and experiment studies, so a set of dies whose profiles can be revised have been designed based on the spring-back theory. Differently from the regular casting press forming mold, the set of dies consist of surface plate and a framework structure of which the assembly and disassembly is easy to made according to the forming dimensional quality. Firstly, the empirical formula has been used to calculate the theoretical die profile value which is mostly determined by the mechanical properties of 316LN. Then, choosing three relevant nearby values to simulate the forming process in which the optimal profile errors of shell must meet the demand of $\pm 2\text{mm}$, Finally, the optimal dimensional value of the molds have been confirmed to guide the forming experiment study. After experiments, the actual profile errors, thickness reduction, maximum spring back, maximum deformation and surface residual stress have been measured. And those data are an accurate indicator of what from the FEA, verifying the reasonability of the manufacture process which can be used to guide the forming of the whole D shape shells.

Key Words: CFETR Vacuum Vessel, Forming Process, Molds Design, Spring-back, Residual Stress

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 83 / 82

Application of EPICS in HL-2A Host Centralized Control System

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In order to improve the capability of Tokamak measurement and control system, centralized measurement and control system with compatibility and continuous upgrade ability is needed to build. EPICS has advantages of safety, easy expansion etc. in this paper, HL-2A host centralized control system based on EPICS is developed and designed. Communication between SoftIO and PLC was realized using the s7nodave device driver module. Through the CSS application development, real-time communication between OPI layer and subsystem has been realized, and the PLC of the subsystems can be integrated to the EPICS control system. In order to manage subsystem of HL-2A host device more intuitive and convenient, the state mode of HL-2A centralized control system is deeply researched based on the theory of CODAC state machine. Standard system states are defined, subsystems are converted to the state in a given order. The state of other system synchronize as the state of a system changes. HL-2A host control system based on the state machine mode is not only easy to centralize management of each subsystem, but also can realize the interlock and protection among subsystems. HL-2A host centralized control system prepare for design on the next generation device host centralized control system.

Eligible for student paper award?:

No

M.OA2: Divertors and High Heat Flux Components / 83

Design and test of W7-X water-cooled Divertor Scraper

Authors: Jean Boscary¹; Henri Greuner¹; Gunnar Ehrke¹; Bernd Böswirth¹; Zhongwei Wang¹; Emily Clark²; Arnold Lumsdaine³; Jörg Tretter¹; Reinhold Stadler⁴; Dean McGinnis³; Jeremy Lore³

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For steady state operation up to 30 minutes pulse duration of the stellarator Wendelstein 7-X, an actively water-cooled divertor will replace the currently installed adiabatically loaded divertor designed for maximal 10 second plasma pulses. Heat load calculations taking into account the effect of bootstrap current have indicated the possible overloading of the ends of the divertor beyond their technological limit. The intention of the additional scraper is the interception of some of the plasma fluxes both upstream and downstream before they reach the divertor surface. To check the effect of the scraper on the divertor for long pulse operation, an adiabatically loaded scraper element will be installed during the phase of the short pulse operation.

Design activities including the manufacturing and testing of prototypes have been carried out to prepare a possible fabrication of the water-cooled scraper. One scraper is made of 24 identical plasma facing components (PFCs). A PFC is 247 mm long and 28 mm wide. It has 13 monoblocks made of CFC NB31 bonded by hot isostatic pressing onto a CuCrZr cooling tube equipped with a copper twisted tape. Due to pressure drop limitation the scraper is divided into 6 parts of 4 PFCs; each part has 4 PFCs hydraulically connected in series by 2 water boxes (inlet and outlet). Individual full-scale prototypes of PFCs have been successfully tested in the GLADIS facility up to 20 MW/m².

This paper discusses the challenges of the design and manufacture of the water box prototypes. The scraper and water boxes have to be integrated in a very limited available space and require a very compact design. Prototypes have been manufactured to select the best technology for the water boxes. The results of the successful HHF testing of a component made of 4 PFCs will be presented. The results of these activities have defined the technological basis for a possible fabrication of the water cooled scraper.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 73 / 84

Recent Development in Structural Design and Optimization of ITER Neutral Beam Manifold

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The Neutral Beam (NB) manifold is a major sub-system of ITER fueling system with complex combination pipes with the aim at distributing the working gases for NB and Diagnostic NB injector. During the final design (FD) phase, NB manifold design has been completed based on configuration management model defined to use in FD. After the FD review, the design of NB manifold suffered several design changes so as to meet the different manifold routing requirements. Additional, structural integrity assessment during FD revealed that the NB manifold design has potential for more robust structure performance, as well as potential for a significant simplification of the support layout by redefining the constraint form of the support and the whole structural architecture. This paper describes the new design of NB manifold based on a more optimized support system. The former complex manifold supports and internal pipe supports have been compacted and replaced with an alternative scheme in order to more effective, decreasing about 90mm of structural deformation. Detailed analyses on internal pipe support layout are dedicated to confirm both the structural reliability and feasibility. Comparative analyses between two typical types of manifold support scheme, with emphasis on space feasibility, embedded plate location and etc., have been performed. All relevant results of thermo-mechanical analyses for different operation scenarios and fault conditions are presented as well as the mechanical behaviors and manufacturing aspects. Future optimization activities are described, which shall give useful information for a refined setting of components in the next phase.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 97 / 85

Research on the Method of Reactive Power Detection for Tokamak Coil Power Supply Based on AC/DC System Active Power Balance

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A new method of real-time reactive power detection is proposed, which aims to be applied in reactive power for Tokamak coil power supply volatility, high random and the contradiction between accuracy and real-time of traditional method. It based on AC/DC system active power balance principle, and considering the electric network voltage sag, distortion and excitation current of transformer. The characteristics of the detection method are high real-time and precision, which not affected by electric network time-varying parameter. The method has been proved correctness and effectiveness by the experimental of reactive power detection for poloidal field power supply in EAST.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 109 / 86

Preliminary Probabilistic Safety Assessment of Tokamak-type Fusion Power Plants In Conceptual Design Stage

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Owing to their inherent safety features and lacking of high level long-lived radioactive waste, fusion power plants have long been expected to be a safe, clean and ultimate solution for human energy crisis. Among all the fusion reactors, the tokamak-type fusion power plant (FPP) with deuterium and tritium fuels is considered to be one of the most promising fusion energy systems. However, there has been no probabilistic safety assessment for severe accidents of this type of FPP. Then how is the radiation risk of a tokamak-type fusion power plant in the viewpoint of severe accidents? This paper is such an effort to assess its radiation risk under the conditions of severe accidents in a risk way.

Since there is no reactor core in fusion power plants, core damage frequency concept from fission nuclear power plants cannot be adopted as the risk metric for fusion power plants. But the frequency of large off-site release of radioactive material could be a possible effective risk metric, as there will also be specific radioactive material release in the accidents of a fusion power plant. According to the recommendations of international atomic energy agency, a large release of radioactive material can be specified in a way as a specified dose to the most exposed person off the site. Therefore, a large release concept could be defined for fusion power plants.

On the basis of this large release concept, preliminary probabilistic safety assessment was applied to the safety design concept of a typical FPP based on the European fusion power plant conceptual study. This complex assessment work is finished with the assistance of reliability and probabilistic safety assessment program RiskA developed by FDS Team. Not only the total large release frequency of a fusion power plant was calculated, but also representative large release accident sequences initiated from specific accident types of a fusion power plant were identified. And their characteristics in happening frequencies, radioactive material release fractions, releasing time were analyzed and compared. Results showed that fusion power plants were not so safe as public's imagination. There are still accident sequences which would arise significant radiation risk, although inherent safety features exist in the tokamak-type FPP.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 33 / 87

Tokamak Size Scaling

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The size scaling is recently developed by Costley, Hugill and Buxton (CHB) for seeking highest tokamak performance within physics limits at fixing the fraction of Greenwald density limit f_{GW} , normalized plasma pressure β_N , and fusion power P_f by scanning device-major-radius, R . The size-cancelling effects of the density limit are found in the fusion triple product of Lawson criterion, $nT\tau_E$ and fusion power gain, Q_f . In CHB scaling, cylindrical geometry plasma is assumed to be in the nearly-full-sized vacuum chamber of tokamaks, together with a minimum of 250 MWt fusion power output in a JET-sized machine [A.E. Costley 2016 Nucl. Fusion 56, 066003]. However the assumption meets the real low-power-gain case of JET experiments at less than 20 MW due to low burn rate of deuterium-tritium (DT) fusion. The compressed plasma is thus suggested for filling the power gaps of existing low-power-gain cases of tokamaks to high-power gain. Existing limitations of EAST tokamak are analyzed for accommodating and simulating the high-performance discharges, including the additional pulsed power suppliers and magnets. Possible operation scenarios of tokamaks are further analyzed for high-gain high-field (HGHF) fusion plasma suggested in [Li. G., Sci. Rep. 5, 15790 (2015)].

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 111 / 88

Latest results from the Hybrid Illinois Device for Research and Applications (HIDRA)

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The Hybrid Illinois Device for Research and Applications (HIDRA) is a toroidal fusion device at the University of Illinois. HIDRA is the former WEGA stellarator that was operated in Greifswald at the Max Planck Institut für Plasmaphysik. The machine is a five period, $l = 2$, $m = 5$ stellarator, with major radius $R_0 = 0.72$ m and minor radius $r = 0.19$ m. Initial heating is achieved with 2.45 GHz ECR heating at $B_0 = 0.087$ T magnetic field, which can go as high as $B_0 = 0.5$ T. HIDRA has the ability to operate as both a stellarator and a tokamak, initially operating in the stellarator mode. The focus of research on HIDRA will be to do dedicated PMI studies using the wealth of knowledge and experience at the Center for Plasma Material Interactions. In early 2016 the first experiments were started to be performed in HIDRA. This paper presents some of the initial results obtained from the machine.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 4 / 89**WCLL BLANKET MODULE STRUCTURE VARIATION INFLUENCE ON NEUTRON ACTIVATION INVENTORIES****Authors:** Andrius Tidikas¹; Gediminas Stankūnas¹¹ *Lithuanian Energy Institute***Corresponding Author:** andrius.tidikas@lei.lt

One of the conceptual designs of the breeder blanket for a European DEMO is the Water Cooled Lithium Lead (WCLL) concept. Design development of the WCLL blanket in recent years has led to evolution of the neutronics model employing the MCNP code, with key changes in radial lengths and thickness as well as dimensional differences in inner structures such as breeding zone, shielding zones and manifold. Furthermore there has also been a significant increase in projected fusion power. Such changes were made with respect to its nuclear, thermohydraulic and thermomechanical performances. In this work neutronic characteristics of WCLL modules were analysed in terms of fusion power increase and dimensional changes of the blanket geometry presented in DEMO 2014 and 2015 models. For comparison, outboard and inboard blanket modules of equatorial region were selected. Investigated neutronic characteristics include activity, decay heat and contact dose rates. Numerical experiment was carried out with MCNP particle transport code and FISPACT activation calculation code using EAF-2010 nuclear data library.

Keywords: nuclear fusion, FISPACT, MCNP, neutron activation, DEMO, WCLL

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

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 71 / 91**An approach to the study of crack initiation at the divertor tungsten target plates of ASDEX Upgrade****Author:** Nikola Jaksic¹**Co-authors:** Muyuan Li¹; Irene Zammuto¹; Albrecht Herrmann¹; Henri Greuner¹; Jeong-Ha You¹; Rudolf Neu¹¹ *Max Planck Institute for Plasma Physics, EURATOM Association***Corresponding Authors:** muyuan.li@ipp.mpg.de, jeong-ha.you@ipp.mpg.de, albrecht.herrmann@ipp.mpg.de, rudolf.neu@ipp.mpg.de, henri.greuner@ipp.mpg.de, nikola.jaksic@ipp.mpg.de, irene.zammuto@ipp.mpg.de

The solid tungsten divertor tiles at ASDEX Upgrade experimental device have been gradually installed as substitution for W coated graphite tiles in 2012.

ASDEX Upgrade is equipped with an adiabatically loaded divertor as a compromise between available heating power, plasma discharge length and heat removal capability of divertor tiles. Accordingly, the design of the new solid tungsten plates has been conditioned by in-vessel surrounding and supporting structure. The full-scale prototypes, dimension of 250x80x15 mm³, have been intensively tested in the high heat flux test facility GLADIS (Garching Large Divertor Sample Test Facility) [1]. The GLADIS heat loading profiles are Gaussian with central heat flux of 10 - 30 MW/m², resulting in an integrated absorbed power of the W tile between 100 and 280 kW. Thus simulating the expected highest power and energy loads in ASDEX Upgrade. The corresponding measured surface temperatures reached values between 1500 °C and 3300 °C. In addition, cyclic loading tests have been performed with 200 cycles at 10.5 MW/m², 3.5 s duration. These applied loads correspond to the expected thermal loading of about 4 years of ASDEX Upgrade operation with approximately 50 high power discharges per campaign.

During the cyclic loading in the GLADIS facility, no crack initiation at the tungsten tiles has been detected. However, after one campaign of AUG operation (about 1200 plasma shots) almost all tungsten divertor tiles exhibit cracks. The inspection of the plasma exposed tiles has revealed 126 tiles with deep cracks. Nearly all of 128 tiles have shown shallow cracks in the high heat load region. Nevertheless, none of these divertor tile damages have caused an operational interruption of ASDEX Upgrade. A comprehensive investigation of the damages has been performed to find out the origin of the crack initiations [2].

This paper is presenting a bundle of numerical simulations of the ASDEX Upgrade solid tungsten divertor tiles on the basis of the theoretical hypothesis for failure of brittle materials. Accordingly, thermomechanical analyses with cyclic loading, simulating both the GLADIS and ASDEX Upgrade load profile have been performed. Additionally an assessment of the crack initiation induced by material fatigue under thermal cyclic load has been studied. Finally, the design optimisation considerations of divertor tiles are discussed.

[1] JAKSIC, N., et al., "FEM investigation and thermo-mechanic tests of the new solid tungsten divertor tile for ASDEX Upgrade", *Fusion Engineering and Design* 88 (2013) 1789–1792, <http://dx.doi.org/10.1016/j.fusengdes>

[2] HERRMANN, A., et al., "Experiences with a solid tungsten divertor in ASDEX Upgrade", *Journal of Nuclear Materials and Energy* (2016)

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 43 / 92

Design of Ground Plane of NSTX-U Ohmic Heating Coil

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A ground plane is a conductive, grounded, electrostatic shield surrounding high voltage insulated conductors. The purpose is to contain the electric field developed by the high voltage conductors within the insulation and to provide a return path for capacitive currents during transients.

A resistive paint was applied to the outer surfaces of the NSTX-U OH coil to form an outer ground plane. It serves to shield the surrounding instrumentation (flux loops, thermocouples, etc.) from electrical noise generated by the switching of the thyristor converters that supply power to the OH coil.

This paper provides a detailed description of the equivalent circuit model of the multi-layered OH coil with the ground plane. An analysis of surface voltage of the OH coil is performed using the PSCAD, a profession transient simulation tool, to confirm that the ground plane performs as required under

normal operation and fault conditions. This work is supported by US DOE Contract No. DE-AC02-09CH11466.

Eligible for student paper award?:

No

W.OA2: Divertors and PFCs: Liquid Metals / 93

Recent results of Li experiments in EAST with W divertor

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In EAST, Li applications with various methods have been systemically developed and significantly improve plasma performance, such as Li evaporation for suppressing impurity radiation and reducing hydrogen recycling [1], Li dropper for 18s ELM-suppressed H-mode discharges [2], Li granule injection for ELM pacing [3], and flowing liquid Li limiter (FLiLi) for improved performance plasma and reduced divertor heat flux [4]. Recently in 2016, Li applications were successfully carried out in EAST in upper-single null configuration using the ITER-like W divertor, and new, exciting results were obtained. A uniform Li coating using new dedicated ovens effectively suppressed W impurities coming into the plasma. Reproducible high performance ELM-eliminated H-mode discharges with high power RF heating have been obtained in the EAST W divertor tokamak, resulting from the continuous real-time injection of a fine Li aerosol into the plasma upper X-point. Moreover, no core impurity accumulation was observed during ELM-free periods. We note that real-time Li aerosol injection appears to promote the growth of the low-n electromagnetic coherent mode (MCM), possibly owing to the increase in Li concentration. Robust ELM pacing with Li granules injection was demonstrated in upper-single null W PFC discharges. It was also observed Li granules injection shifted the density profile outward, then change the characters of edge fluctuation. During 2016 Flowing Liquid Lithium (FLiLi) limiter experiments, some engineering significant improvement were realized, including improvement of liquid Li surface coverage uniformity (~80%), without no obvious limiter surface damage and no Li passive bursts. Moreover, improved plasma performance during transient ELM-free H-modes, with a strong increase of WMHD and H98 were demonstrated, along with full-field ohmic H-mode. Overall, the Li coating provided an excellent wall conditioning in W divertor, facilitating the 62s EAST H-mode. Those new results of Li applications with W PFCs wall in EAST would significantly extend the Li potential applications in future reactors.

Reference

- [1] G.Z. Zuo et al., Plasma Phys. Contr. Fusion 54 (2012) 0115014
- [2] J.S. Hu et al., Phys.Rev. Letts. 114 (2015) 155001
- [3] D.K. Mansfield et al., Nucl. Fusion 53 (2013) 113023
- [4] J.S. Hu et al., Nucl. Fusion 56 (2016) 046011

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 11 / 95

Current profile measured by the motional Stark effect polarimeter in the HL-2A tokamak

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The safety factor and current density profiles play a very important role in understanding magneto-hydrodynamics and micro-instability. Motional Stark effect (MSE) is one of the most powerful tools to measure the current density. A 4-channel MSE polarimeter based on dual photo-elastic modulators (PEMs) has been developed in the HL-2A tokamak. For each channel, 6 1-millimeter silicon fibers are applied. And off-the-shelf avalanche photodiode detectors with frequency band of 250 kHz are adapted due to its quantum efficiency up to ~83% at 660 nm. The beam emission spectra are filtered by a monochromator; and the filter is controlled by an absolutely calibrated rotator, which can change the tilting angle of the filter with velocity of 720 degree/s, corresponding to the wavelength change of 288 nm/s with the filter. The rapid angle change of the monochromator enables the wavelength to be swept during the discharge. The accuracy of the MSE can be up to $\pm 0.15^\circ$ in the calibration experiments.

On HL-2A, the motional Stark effect is rather weak [1]. During the pilot experiment, the pitch angles of magnetic field are obtained for 3 spatial channels covering 10 cm along the major radius with time resolution of 5 ms. The profiles of current density and safety factor are obtained by the Current Profile Fitting (CPF) code, as shown in Figure 1. The q profile is monotonic, and the minimum q value is around 0.7. And the position of the q=1 surface consists with the sawtooth inversion radius measured by ECE.

[1] D. L. Yu et al., Rev. Sci. Instrum., 85, 053508 (2014).

Eligible for student paper award?:

Yes

T.OP1: Power Supply Systems / 96

Development of HL-2M power supply system

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Co-authors: Weimin Xuan ; Lieying Yao ²; Huajun Li ²; Xiaolong Liu ²; Jianfei Peng ²; Haibing Wang ; Weibin Li ²; Yali Wang ²

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Southwestern Institute of Physics (SWIP) is establishing a new tokamak (HL-2M) which needs about doubled power capacity and energy than its existing tokamak (HL-2A). The HL-2M power supply system shall meet these requirements, make full use of existing power supply equipment and provide the operation compatibility for both HL-2A and HL-2M.

Besides three existing flywheel six-phase motor generators (MG), two 90 MVA and a 125 MVA, with horizontal configuration, another 300 MVA MG with vertical shaft, which is the first vertical type pulsed MG in China, is designed and manufactured. The weight of the new MG is about 800 t and the rotor is about 400 t. Its released energy in one shot can be 1350 MJ which is the sum of three existing horizontal MGs.

A new toroidal field (TF) filed power supply is developed to provide 1660V/140kA for HL-2M. Eight

paralleled diode rectifiers can be divided by two parts and connected in series to provide compatibility of 45kA with higher voltage for HL-2A TF coil. The output of rectifiers is controlled by adjusting the excitation of motor generator. Four sets of three-winding transformer with extended delta $\pm 15^\circ$ are adopted to provide better impedance symmetry than delta/star configuration.

Central solenoid (CS) power supply is the most challenging one in HL-2M. It is a four-quadrant thyristor converter ($\pm 750\text{V}/\pm 220\text{ kA}$) with circulating current mode. Special design is considered both on the transformer and converter configuration. A bi-direction bypass unit which consists of 120 thyristors is developed to protect the system in case of faults. There are 16 poloidal power supplies (PF) in HL-2M, which contribute not only equilibrium but also volt-second. Therefore most of them are four-quadrant thyristor converters.

Besides the main circuit, there are also improvements on the control system, PLC and data acquisition. For example, the converter is fired by an integrated real time control system with reflective memory communication with central control. A synchronization signal processor based on a digital filter is developed, to obtain clean and accurate reference from the varying frequency and distorted output voltage of the motor generator.

Eligible for student paper award?:

No

W.OP3: Blankets and Tritium Breeding: Solid Breeders / 98

Progress in design activities related to the water cooled breeder blanket for CFETR Phase-I

Author: Songlin Liu¹

Co-authors: Xuebin Ma¹; Lei Chen¹; Xiaoman Cheng¹; Kecheng Jiang¹; Xiaokang Zhang¹; Dingyu Lao¹; Yuohua Chen²; Kai Huang¹; Qingjun Zhu¹; Wanqing Wang¹; Min Li¹

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The Chinese Fusion Engineering Testing Reactor (CFETR) is the next device in the roadmap for the realization of fusion energy in China. It will be operated in two phases. The fusion power is up to 200MW for Phase-I and will be over 1GW for Phase-II. The water cooled breeder blanket (WCCB) is one of three candidates.

The WCCB modules surrounding the plasma and its primary heat transfer system (PHTS) have been designed for Phase-I. The blanket module features are that the cooling plates and the breeder zone are parallel to the first wall (FW); purge gas is directed in the toroidal direction to reduce its pressure drop; the mixed breeder of Li_2TiO_3 and Be_2Ti are chosen as tritium breeder and primary neutron multiplier, a bit of Be as supplement of multiplying neutrons, RAFM steel as structural material, tungsten as armor material of the FW. Pressurized water of 15.5MPa is chosen as coolant with 285°C inlet/325°C outlet. The performance analyses related to neutronics, thermohydraulics, and mechanics have been performed for the whole three-dimension geometry of typical blanket module. In order to support the design of the WCCB, the flow characteristic of purge gas in the WCCB mixed pebble bed is studied. The safety issues related to tritium, water activation, and accidents are also assessed and some mitigation measures are recommended. In addition, the fabrication process of the FW with tungsten armor of the WCCB is being developed. In this paper, the above-mentioned design activities will be introduced.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 12 / 99

Lessons learned on design, manufacturing and commissioning of IRVIS endoscopes prototypes for W7-X divertor temperature monitoring

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The Wendelstein 7-X fusion experiment at Max-Planck-Institut für Plasma Physik (IPP) in Greifswald produced its first hydrogen plasma on 3rd February 2016. This marks the start of scientific operation. Wendelstein 7-X is to investigate this configuration's suitability for use in a power plant. In order to allow for an early integral test of the main systems needed for plasma operation (magnet system, vacuum, plasma heating, control and data acquisition, etc), one of the five divertor unit (module 5) and most of the carbon tiles covering the wall protection elements are being installed before the next experimental campaign (OP1.2). For the later operation phases, the heat fluxes coming from the plasma will be distributed over an area provided by the plasma facing components (i.e. divertor target plates, baffles). An important diagnostic for W7-X will be thermography systems monitoring the surface temperature of the divertor target plates by collecting and processing infrared (IR) and visible (VIS) light from the divertor region of the plasma. For this purpose the company Thales SESO has been assigned to design, build, test, deliver and install two prototypes of IRVIS (InfraRed-VISible) endoscope systems for the divertor of the W7-X Stellarator. Thermography is part of the operational and protective divertor diagnostics and has to detect signals indicating anomalous operation scenarios. The design of the horizontal and vertical target plates and the baffles in the divertor should keep the local power load below 10 MW/m². The IRVIS endoscope systems are designed to operate under heavy-duty conditions.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 98 / 100

Experimental Investigation on the Second Commutating Process of a Quench Protection Switch

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The quench protection switch (QPS) is an indispensable component to ensure the safety of the magnet coils of a superconductive tokamak when a quench happens. The two most important functions

of a QPS are to carry high direct current during normal operation and to interrupt the high direct current when a quench occurs. In this paper, the second commutating process of a QPS based on artificial current zero is investigated. In this process, the current, which has already transferred from the by-pass switch to the main circuit breaker (vacuum circuit breaker), is forced to commute from the vacuum circuit breaker to the dump resistance by the counter current. A LC oscillating circuit is applied to generate oscillating current to simulate the direct current near its peak which is in the range of 4-20kA. The counter current with frequency of 500Hz and 1000Hz is provided by a pre-charged capacitor bank. The equivalence of the interrupting process between practical direct current source and LC oscillating source is analyzed. The vacuum interrupter of the vacuum circuit breaker adopts a pair of contacts generating transverse magnetic field. The evolution of vacuum arc in the interrupting process is investigated by a high-speed camera with exposure time of 2 μ s. The experiment results indicate that the initial process and the motion of the vacuum arc before injecting the counter current have crucial impacts on the interruption performance.

The research was supported in part by the National Magnetic Confinement Fusion Energy Research Project under Project No. 2015GB121005, and in part by the National Natural Science Foundation of China under Project Nos. 51322706 and 51325705.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 5 / 101

Transverse velocity effect on Hunt's flow

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The study of magnetohydrodynamic (MHD) flow is important in the application of the liquid metal blankets in thermal nuclear fusion reactors. Hunt presented an analytical solution of the conducting fluid flow in a rectangular duct under a uniform transverse magnetic field with insulating walls parallel to the magnetic field and thin walls with arbitrary conductivity perpendicular the magnetic field. The case is called Hunt's case II, which is recommended as a benchmark to verify and validate MHD codes related to fusion applications. Hunt's analytical solution is based on two dimensional fully developed laminar flow assumption, which means that the velocity vector has only flow direction component as a function of planar space. However, the transverse velocity component vertical to the streamwise and magnetic field direction in three dimensional numerical simulation increases with the increasing of the Reynolds number and redistributes the streamwise velocity even the flow is laminar.

Hunt's case II with the same Hartmann number ($Ha=500$), wall conductance ratio ($c=0.1$) and wide range Reynolds number ($2000 \leq Re \leq 10000$) have been simulated using three dimensional MHD solver developed in OpenFOAM environment to investigate when the transverse velocity must be considered. The streamwise velocity and the pressure gradient obtained from numerical simulation are compared with Hunt's analytical solutions, which are set as the standard results. If the relative difference of the maximum velocity or pressure gradient is more than 5%, it is defined that the two dimensional assumption breaks down. The results show that the relative difference of the velocity and the pressure gradient increases when $Re \geq 4000$. The relative difference of the velocity is 9.551% when $Re=8000$. The relative difference of the velocity and the pressure gradient is 14.152% and

6.229% respectively when $Re=10000$. Numerical simulation shows that the dimensionless transverse velocity normalized increases and rises from the order of 10^{-5} to 10^{-3} .

The effects of the Hartmann number on the transverse velocity has been investigated by simulating Hunt's flow at $Re=4000$, $C=0.1$ and $50 \leq Ha \leq 1000$. It shows that the relative difference of the maximum velocity is more than 20% when the Hartmann number is less than 100, which indicates that the dimensionless transverse velocity is the order of 10^{-3} .

Finally, the transverse velocity effect as a result of the wall conductance ratio is simulated. The difference of the pressure gradient is lower than 5% for $c=0.01$. When $Re/Ha > 12$, the difference of the maximum velocity increases linearly with the Re/Ha increasing for any wall conductance ratio.

In conclusion, the comparison of the three dimensional numerical simulation and the two dimensional analytical solution of the Hunt's case II shows that the transverse velocity is not negligible when Reynolds number is more than 4000. The transverse velocity is also influenced by the wall conductance ratio and the Hartmann numbers. As a result, if the dimensionless transverse velocity is over the order of 10^{-3} , the two dimensional assumption becomes invalid and the analytical solution is no longer suitable for high Reynolds number MHD duct flow validation.

Key words: Transverse velocity, Magnetohydrodynamic, Hunt's flow, numerical simulation, three dimensional

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 31 / 102

Final Design and Fabrication of the TDU Scrapper for Wendelstein 7-X

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A divertor "scraper" has been designed to protect weakly-cooled regions of the Wendelstein 7 X (W7-X) divertor targets from overheating under certain steady-state conditions [1]. The scraper is a limiter-like component with a plasma-facing profile geometry that was numerically designed to shadow those vulnerable target regions under such conditions. The "TDU" scraper is an inertially cooled component designed to test the protective function of the scraper, and its impact on particle pumping, under pulsed operating conditions during the OP1.2 campaign in 2017-18. The TDU scraper design matches the steady-state profile geometry within 0.20 mm, but was simplified to reduce cost while satisfying requirements for pulsed W7-X operation with 80 MJ total plasma energy input and pulse intervals up to 20 minutes, constrained by the very limited space available near the divertor. The scraper units are instrumented with in-situ Langmuir probes, thermocouples, and a pressure manometer to support the test program. These test divertor unit scraper elements "TDU-SE" components will be located in two of the ten half modules during W7-X OP1.2 operation. This paper describes the design considerations and fabrication of these scraper elements for Wendelstein 7-X.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 99 / 104

Application of the voltage control mode of second-generation EAST active feedback power supply

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The ability of magnetic confinement to plasma can be improved by elongating plasma cross-section in EAST (Experimental Advanced Superconducting Tokamak). But elongated plasma has vertical displacement instability, without control, plasma will dash against wall of vacuum vessel and disrupt, that will cause failure of plasma discharge. So feedback control system is needed to restrain plasma vertical displacement. PCS (Plasma Control System) detects the vertical displacement of the plasma and calculates the value of signal sent to power supply, the signal is real-time tracked and linearly amplified to generate a fast-changing magnetic field, which will suppress the vertically unstable displacement of the plasma.

The analog control was adopted in the first-generation active feedback power supply, which worked in current tracking mode. The conventional proportional regulator which guarantees satisfactory control accuracy of the output current was adopted.

DSP (Digital Signal Processing) was adopted as the main control chip. To achieve the maximum current's rising rate, second-generation EAST plasma vertical displacement active feedback power supply applies voltage control mode while retaining the first-generation current tracking mode. Its average output voltage value is linear to given voltage signal. For a given voltage signal of 10V, the power supply outputs 1600V, and -1600V corresponds -10V. Compared with current mode, voltage mode achieves a significant increase in rising rate of load current. However PCS cannot be fully counted on to detect the real-time load current and change the polarity of the given signal before the current exceeding its limitation, so power supply system itself must possess perfect over-current protection function.

The driving signal is blocked when the output current reaches the protection threshold value and resumed after falling below a certain set value. In this status, the stored energy within the load coil inductance can only be released through the inverse parallel diode of the IGBT (Insulated Gate Bipolar Transistor) to storage capacitors on DC side, which will lead to continuous increase of DC voltage. When the power supply outputs high voltage, the accompanied frequent over-current protection will return the power to DC side, which leads to over-voltage protection. To solve this problem, the current limiting control mode is adopted. The current limiting mode will stay till the polarity of input voltage signal changes.

In 2014 EAST experiment, voltage mode was applied to plasma vertically unstable displacement by the second-generation active feedback power supply. In 52444th experiment, active feedback system exhibits great control ability to vertical displacement of plasma. Even plasma reaches vertical displacement of 4.6cm and growth rate is 530/s, the active feedback system is still able to pull it back to equilibrium position, while the first-generation active feedback power supply can only deal with 1.9cm of plasma vertical displacement and growth rate is 150/s at most.

Through exploration of voltage mode, combination of voltage open loop control and current limiting control is present and the control effect was verified by EAST experiment, which will provide new idea to control vertical displacement of the plasma.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 32 / 105

Numerical Simulation Research on the New Design Scheme of the EAST Divertor using Multi-physics Coupling Method

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The surface thermal flux density on the EAST divertor would reach 10MW/m² due to improvement of heating capacity. The lower divertor as one of the main in-vessel components in EAST must be updated to fit the future experiments for achieving high performance and long pulse plasma, which aims to bridge the gaps between the high performance experimental device EAST and the Chinese Fusion Engineering Test Reactor (CFETR).

Now, some new design schemes of the lower divertors have been proposed and studied. All the possible loads including thermal flux, electromagnetic force, eddy electromagnetic moment and so on will be considered by applying the multi-physics coupling simulation method, so the simulation results would be closer to actual condition. The design schemes which couldn't satisfy the requirements will be optimized until success.

The research efficiency could be improved and the research costs could be decreased greatly by the multi-physics coupling simulation method. All the research results will be provide valuable references to the design of the divertor or other in-vessel components on EAST and CFETR.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 104 / 106

Progress of Auxiliary Systems for Linear IFMIF Prototype Accelerator (IFMIF)

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Progress of Auxiliary Systems for Linear IFMIF Prototype Accelerator (LIPAc)

Authors

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The International Fusion Material Irradiation Facility (IFMIF) aims at qualifying and characterising materials capable to withstand the intense neutron flux originated in the D-T reactions of future fusion reactors thanks to a neutron flux with a broad peak at 14 MeV capable to provide >20 dpa/fpy on small specimens also qualified in this Engineering Validation Engineering Design Activity (EVEDA) phase. All its broad mandate has been successfully achieved, the only pending, is the validation of its Linea IFMIF Prototype Accelerator (LIPAc) with its Auxiliary Systems.

The validation of LIPAc will be achieved in this on-going phase until December 2019 with the operation of a deuteron accelerator at 125 mA CW mode and 9 MeV, which is presently under installation and commissioning in Rokkasho (Japan). The successful operation of such a challenging plant, demands careful assessment of its auxiliary systems, that holding adequate redundancies will allow

the target plant availability. The target availability of LIPAc was considered top priority even due to the inherent administrative difficulties of an “in-kind” project.

LIPAc, the Linear IFMIF Prototype Accelerator presents a broad spectrum of ancillary equipment to optimize its operational beam time.

A description of the Nuclear HVAC of IFMIF has already been reported [1].

The present paper describes the auxiliary systems of LIPAc, (and their construction status) among which we address the Cryoplant System (Cryo), the Heating Ventilation & Air Conditioning (HVAC), Electrical Power Supply (EPS), the Service Water System (SWS), the Service Gas System (SGS), the Heat Rejection System (HRS) and the Fire Protection System (FPS).

[1] G. Pruneri et al., Design principles of a nuclear and industrial HVAC of IFMIF, Fusion Engineering and Design 103 (2016) 81–84

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Topic: Project management, systems engineering

Oral or poster preference: Poster

Eligible for student paper award?:

No

T.OA1: Diagnostics and Instrumentation I / 107

Prototype manufacturing and testing of metalized ceramic printed circuit boards for ITER Bolometer cameras

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The ITER bolometer diagnostic will be based on 110 five-channel metal resistive sensors distributed all around the vessel [1]. In order for the diagnostic to meet its operational and programmatic goals, a mechanically stable and reliable electric connection is required. Spring loaded connection of the sensor or crimped wires to connect the external cables, as used in current day fusion experiments, are not suitable to operate under ITER’s challenging nuclear and thermal loads. In addition, the design must not exceed its space envelope and an efficient way how to customize the sensor position according to the final blanket module positions has to be demonstrated. Excellent thermal conductivity in these internal components is also a requirement in order not to overheat the sensor.

In this paper, a design solution based on metalized ceramic aluminum nitride substrates providing the electrical interface between the internal bolometer sensor and the externally connecting macroscopic MI cables, is proposed. The substrates, also referred to as printed circuit boards (PCB), can be manufactured having a complex 3-D shape and can be coated on multiple sides with micrometer thick conductive tracks and pads.

It is shown, that the sensor can be supported mechanically with an integrated design small enough to fit into the tight space envelope reserved for the ITER bolometer cameras. A solution, to allow flexibility in sensor positioning during assembly, will be explained. Investigations into bonding and micro welding techniques to provide a reliable electrical connection as well as the possibility to integrate a remote-handling compatible connection will be discussed.

Prototypes based on a simplified PCB design have been manufactured by micro-dispensing, laser etching and laser activation in order to validate these technologies for the demanding ITER environment. To determine the exact specifications and design constraints for the final electrical interface,

these simplified PCBs contain vias, bond pads coated with different material combinations (Au, Pt, Cu) and tracks running over 3-D shaped surfaces.

Test results on mechanical stability and electrical properties of the different simplified PCBs before and after thermal cycling are discussed together with the analysis of the achieved electrical track widths and thicknesses and their impact on the diagnostic performance. Moreover, conclusions and considerations on cost-effective manufacturing will be presented. The paper concludes with an outlook describing the preliminary bonding specifications and challenges for the internal bolometer sensor and the external MI cables.

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Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 60 / 108

Modeling of advanced nuclear fuel cycles incorporating hybrid fission/fusion devices.

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Two of the main cited flaws of nuclear fission power to label it as a non-sustainable energy source are linked to the nuclear fuel cycle: one is the fuel availability, and the other is the radioactive spent fuel legacy. The incorporation of fast neutron systems to the fuel cycle can help reduce this two problems, by breeding additional fissile material (thus extending the nuclear resource's lifetime) and by eliminating minor actinides present in the spent fuel that contribute to reducing its radiotoxicity by many orders of magnitude (thus greatly reducing the spent fuel legacy issue). Traditionally, the fission community has explored this alternative via fast breeder reactor designs and their incorporation into the fuel cycle, but the use of fusion-based fast neutron sources needs consideration as well. Jointly, UT Austin and IPN have developed a nuclear fuel cycle modeling platform, which can perform detailed neutronic calculations of both thermal fission and fast fission/fusion systems, and allows material exchange between them at the end of each burn cycle. A Compact Fusion Neutron Source (CFNS), a simplified spherical tokamak design developed at the University of Texas at Austin, generates the neutrons in the fission/fusion device. The CFNS has around it an annular space where zones that contains fresh fertile material can breed fissile material, while other zones may contain spent fuel material that can be "rejuvenated" (i.e. breed additional fissile material) and reduce its radiotoxicity by destroying the minor actinides with fast neutrons. Results from the use of this platform to analyze the self-sufficiency of Th/U and U/Pu fuel cycles with and without reprocessing stages, in particular with regard to neutron economy in the hybrid system, will be presented in this paper.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 110 / 109

Safety impact of the Be-steam reaction during in-box LOCA on the WCCB blanket for CFETR

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The Water Cooled Ceramic Breeder (WCCB) blanket is one of the blanket candidates for Chinese Fusion Engineering Test Reactor (CFETR). In case of in-box LOCA, the cooling channels inside the blanket module are broken, causing leakage and vaporization of the high temperature and high pressure water coolant into the Beryllium pebble beds which are used as neutron multiplier. Then the Be-steam reaction will take place and impact the safety of the blanket system, as well as the fusion reactor. Therefore the safety characteristics of the WCCB blanket during in-box LOCA should be investigated to prevent serious damage. In this paper, RELAP5 which is a system analysis code, is employed to model the blanket modules and the Primary Heat Transfer System (PHTS) of blankets. And the in-box LOCA is simulated by RELAP5 to figure out the transient response of the blanket system with different break areas. Then parameters, such as the steam flow rate at the break, pressure and temperature are transferred to ANSYS CFX for the simulation of the Be-steam reaction in the Be pebble beds. The results show that the impact of hydrogen production is limited. And the system can response in time to mitigate the consequences. Some improvement measures for the WCCB blanket system are recommended.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 105 / 110

PPPL's Project Management Office: Work Planning System

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Princeton Plasma Physics Laboratory (PPPL) is a U.S. national laboratory under contract by the Department of Energy to conduct plasma physics and nuclear fusion research. With an annual budget of approximately 100 million dollars, the laboratory supports projects in various divisions such as: experimental physics, engineering, operations, IT, and facilities and infrastructure. PPPL's Project Management Office (PMO) is responsible for the oversight and implementation of all projects at the laboratory by using an online Work Planning System database.

The Work Planning System implements project management and integrated safety management tools into a comprehensive checklist for projects. It captures the scope of the project, determines project risks, interfaces, support groups, and reviews all engineering work to combine these categories into a holistic approach for the life cycle of a project. Moreover, it guidelines risk management processes and focuses on project roles and approvals.

The Work Planning System is important because it communicates project requirements and progress across a large board of people, consisting of engineers, supervisors, and directors. Monthly reviews are held for all active projects listed in the system to ensure accuracy and compliance, as well as

provide oversight and feedback. This type of Work Breakdown Structure (WBS) is used by everybody at the laboratory; however, it is most important to the engineering staff working on large jobs that require multiple personnel and have a significant cost and time involvement.

In this paper, we will further explore the key functions of the PPPL's Work Planning System, highlight its importance in the project's life cycle, and discuss the development of a planned upgrade.

Eligible for student paper award?:

No

W.PLN: Plenary W / 111

Status of IFMIF Project: is it still talking about IFMIF like talking of Alice in Wonderland?

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The efforts towards the realization of a fusion relevant neutron source facility for fusion materials testing is four decades old. Fusion and fission materials research have always presented synergies, but whereas experimental fission reactors were available since the 60s supporting the development of commercial fission reactors, there is no available facility providing the 14.1 MeV mono-energetic neutrons of DT fusion reactions. This lack led to assess the degradation of structural materials exposed to fusion neutrons through the combination of results from fission reactors, spallation sources and ion implantation, but uncertainties of the degradation at high doses remain. Many ideas have been attempted to have a fusion relevant neutron source, some of these return in an iterative manner, unfortunately simplistic solutions are not available. The US took the lead in late 70s proposing the Fusion Materials Irradiation Test facility, FMIT, a project active until middle 80s; we learnt that our technology was not ready at the time to have neutrons through Li(d,n) reactions. Accelerators technologies have matured enormously these last 30 years with the frontier of MW beam power trespassed this decade in the SNS and 5 MW to be reached early next decade in the ESS. Efforts in Japan and US continued in a timid manner until a consensus was reached in 1994 among Europe, Japan, the Russian Federation and the US to jointly work towards the International Fusion Materials Irradiation Facility. In 2007, EURATOM and Japan signed the Broader Approach Agreement in the field of Fusion Energy Research, which included IFMIF/EVEDA, what stands for Engineering Validation and Engineering Design Activities. The on-going success of this project, with the construction of validating prototypes of the main elements of IFMIF, allows moving towards the construction of a Li(d,n) fusion relevant neutron source for a marginal cost of a fusion reactor, fulfilling the expectations of the world fusion roadmaps to counting with 14 MeV neutrons, with suitable fluxes for fusion materials testing, by the 2nd half of next decade.

Eligible for student paper award?:

No

M.OA3: Inertial Fusion Engineering and Alternate Concepts / 112

Properties of a Clean and Economic Boron Laser Fusion Reactor

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Fusion reactions of protons with the boron isotope 11 (HB11) were considered as extremely difficult and impossible for a power reactor. This changed by several orders of magnitudes using picosecond (ps) lasers with powers >petawatt (PW) igniting fusion in a non-thermal way by direct conversion of laser energy into ultrahigh acceleration of plasma blocks [1]. The HB11 reaction produces primarily only clean helium without nuclear radiation problem. The design of a new kind [2] of a fusion power reactor, (see Fig. 19 of Ref. [3]) contains a reaction unit in the center of the reactor sphere with a cylindrical solid stoichiometric hydrogen-11boron fuel (see Fig. 10 of Ref. [3]). The unit is charged at about -1.4 Megavolt within the reactor sphere and the fusion reaction is generated end on at the fuel cylinder by a 30 kJ laser pulse of ps duration. The 2.9 MeV helium (alpha particles) convert their energy into electricity when moving against the wall of the reactor. At a one Hertz operation rate, the current of 780 Amps is converted into ac three-phase electricity resulting in power generation on a profitable level [3].

The fusion reaction in the cylindrical fuel is trapped by a magnetic field by a 4.5 kilotesla magnetic field for one nanosecond. The trapping field is generated for one nanosecond by the capacitor laser driving device following Fujioka et al. [4]. The conditions for sufficient magnetic trapping of the HB11 reaction for binary reactions in a fuel cylinder of 1 and of 0.2 mm diameter are confirmed by hydrodynamics [2][3] and extended to experimentally confirmed avalanche reactions [5]. Results on physical solutions are reported focusing on direct drive ignition conditions and the theory of avalanche reactions by elastic nuclear collisions [6]. This was elaborated on block ignition by laser pulses of >30kJ-ps producing >GJ energy in the 2.9 MeV alphas. It is estimated that the available 10PW-ps pulses per minute [7] are developed within reasonable time to the 30PW-ps laser pulses for one Hz operation for the new reactor type.

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Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 13 / 113

The measurement of visible bremsstrahlung emissivity profiles on HL-2A

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Reliable experimental determination of the mean effective charge (Z_{eff}) is of great importance for the impurity control in high temperature plasma. The Z_{eff} is usually calculated using data from line integrated bremsstrahlung measurements, electron density and temperature profiles, and the plasma geometry. The visible bremsstrahlung emissivity profiles are achieved with the spatial resolution of 1 cm on HL-2A tokamak for the first time. Light-absorption panels that face the telescope are installed on the inner vessel wall, aiming at reducing the influence of wall reflection on the bremsstrahlung measurements. High-resolution spectrometer is used to select the impurity-line free region instead of interference filters, owing to the fact that in discharges with impurity injection or auxiliary heating power injection, some unpredicted impurity lines may invade this region, giving rise to a deviation of profile measurements. The line integrated bremsstrahlung emissivity profile evolutions are obtained under different discharge conditions, such as NBI and/or ECRH injection, L-H mode transition, and impurity injection, etc. It is found that compared to the center-peaked profile shape in Ohm discharges, NBI power gives a different profile shape with another high peak in the vicinity of $r=26\text{cm}$.

Eligible for student paper award?:

No

R.OP4: Stellarators / 114

Prospects for stellarators based on additive manufacturing

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The work studies the potential and limitations of additively manufactured (AM) stellarators deduced from recent experimental works.

High geometric complexity with high accuracy required for stellarator components (so high cost) delays the construction of new stellarators and the related research. Additive manufacturing (AM) is particularly suited to high geometrical complexity at moderate cost. Thus, exploring whether AM might provide advantages to the fabrication and design philosophy of stellarators appears valuable.

Recent experimental works have explored: i) the dimensional accuracy of several coil frame supports and coil casings AM in plastic, ii) a modular coil frame support produced by combination of an AM structure and casting, iii) fibre-resin casting and a liner for a vacuum vessel sector, and iv) an AM full sector of a small stellarator under vacuum and e-beam mapping. The results from such assays are reviewed and integrated in the paper.

Marginally acceptable (0.1% accuracy for 68% of measurements, $1-\sigma$) dimensional accuracy of AM in plastics has been measured. The low stiffness and strength of AM plastics is tackled with the use of fibre-reinforced resin cast in AM hollow truss structures (3Dformwork technique). A copper liner embedded in an AM shell and resin was satisfactorily assayed and, the present high cost of direct metal AM is circumvented by metal electrodeposition on fibre-resin sectors of vacuum vessel (under development). Additionally, in particular for stellarators, the AM inexpensive complexity allows: i) extra design freedom like 3D-printing of features for conductor crossovers or elements for simple and accurate positioning of the stellarator sectors, ii) the use of numerous grooves for the quick winding of numerous and highly contorted coils (so, lower magnetic ripple and wider distance plasma-coils), iii) all stellarator legs produced on a single piece which decrease assembling time. Nevertheless, AM still is burdened with: accuracy limited to 0.1% under the best commercial plastic printers, small size and expensive direct metal commercial AM, low stiffness and strength of plastics or extra complexity of casting or laminating composite resins, long-term dimensional instability of plastics and

composites, and size of current commercial plastic printers limited to less than 1-2 m side length. Additive manufacturing, particularly combined with other traditional fabrication methods, has proved to be at the verge of achieving appropriate performance for small and medium size stellarators. It might accelerate the production of a diversity of stellarator configurations and coil dispositions, of importance for plasma research and for faster evolution of the stellarator research line.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 61 / 115

ENGINEERING METHODOLOGY TO PROVIDE INTEGRITY OF THE ITER PORT PLUG ON-BOARD COMPONENTS DYNAMICALLY RESPONDING TO PLASMA TRANSIENTS COMBINED WITH SEISMIC EVENTS

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Forschungszentrum Jülich and partners have been developing the ITER upper port plug diagnostic system (cCXRS) that is to transmit the visible light from the plasma to the end diagnostic via optical mirrors.

Each port plug (PP) and its on-board components should withstand severe loads due to the plasma transients when the eddy currents and electromagnetic (EM) forces occur in the PP massive structures and its on-board components. The worst loading case can be found, and the forces be applied as a time-history loading to the PP mechanical model.

In addition, huge transient eddy current and shock dynamic EM forces occur in the vacuum vessel (VV). In response to this loading the VV vibrates, thus exciting the PP and, consequently, its on-board components. ITER Organization has studied a wide range of plasma transients to provide the enveloped Floor Response Spectra (FRS) for different VV locations.

The main computational problem is a reasonable superposition of a deterministically calculated time-history PP response to the applied EM forces with the PP response to the VV excitations that are specified as a FRS at the port stub (port attachment to VV) when only the maximum values of the structure response are calculated over a range of frequencies. On top of this, the EM loads should be combined with the seismic ones which are also specified as a FRS.

This paper considers a potential methodology for combining plasma transients with seismic events. It is not yet intended for design purposes. The paper presents a step-by-step numerical modeling of the upper PP hosting some representative cCXRS component. Approaches to calculate the EM forces in the PP and its on-board components with the use of the dedicated global EM ITER model and to perform a subsequent structural dynamic analysis using a dedicated PP model are presented. The response spectrum analysis (RSA) of the PP and its on-board component that are excited via the VV due to the plasma transients and seismic events are then discussed. The challenge of combining closely spaced modes is highlighted. The approach to superimpose the time-history and RSA results is represented. A conservatism of the proposed approach, its requirements and merits are discussed. The technique proposed herewith is especially demanded when the dynamic behavior of the on-board component is a key feature of its design. This methodology gives a direct and transparent engineering way to design and estimate mechanical strength of the PP on-board components. The analysis uses reliable port stub FRS input and does not depend on spectra-to-spectra recalculation procedure (from port stub to component attachment) that is well established for the seismic-type response spectra but needs to be validated for the FRS due to plasma transients.

This work was supported by Fusion for Energy (F4E) under the Framework Partnership Agreement F4E-FPA-408 (DG). The views and opinions expressed herein do not necessarily reflect those of F4E.

Eligible for student paper award?:

No

R.PLN: Plenary R / 117

Prospect towards steady-state helical fusion reactor based on progress of LHD project entering the deuterium experiment phase

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Large Helical Device (LHD) is one of the world-largest superconducting fusion experiment devices, having demonstrated its inherent advantage for steady-state operation since the start of experiments in 1998. LHD has also demonstrated reliable operation of the large-scale superconducting magnet system for more than eighteen years. Development of the challenging heating systems, such as negative-ion-based NBI, high-power and high-frequency ECH and steady-state ICH, have led to achieving high-performance plasmas, individually, with T_i of 8.1 keV, T_e above 10 keV, volume-averaged beta of 5.1 %, and steady-state operation with the world-record total injected energy of 3.36 GJ.

LHD has progressed to the next stage, that is, the deuterium experiment starting in March 2017, which should further extend plasma parameters towards reactor-relevant regime. For establishing firm basis for designing steady-state helical fusion reactor, advanced physics research, such as on isotope effect, energetic particle confinement, and plasma-wall interaction, will be intensively performed in the deuterium experiments. In an engineering aspect, the upgrade of NBI system has been executed in preparation to the deuterium experiment, and it should contribute to future NBI development for fusion reactors including ITER. For enhancement of the particle control, the closed divertor system has been installed with pumping capability. Diagnostics for neutron measurements are newly developed and installed for the deuterium experiment.

Aligned with all the progress of LHD project in terms of engineering and physics aspects, the conceptual design activity of the LHD-type helical fusion reactor, FFHR-d1, has been programmatically conducted. In parallel to the design study, engineering R&Ds for the component development have been performed, including those based on employing challenging ideas such as high-temperature superconductor, liquid metal ergodic divertor, and molten salt breeder blanket.

The present status of LHD project entering the deuterium experiment phase is overviewed with an emphasis on the engineering aspects, and then the engineering R&D activities towards steady-state helical fusion reactor are presented.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 74 / 118

Design of a local oscillator for the 2.45GHz/4MW LHCD system on EAST

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The paper describes the design process and experimental validation of a local oscillator for the lower hybrid current system(LHCD) on EAST. The local oscillator is designed to provide original RF energy for the whole LHCD system, which plays an important role. The local oscillator must be of high spectral purity and stability. Only phase noise is better than $-90\text{dBc}/10\text{KHz}$ can satisfy the requirements of the LHCD system. The local oscillator consists of the phase locking loop(PLL), the PIN switch, the regulable attenuation, the amplifier, the coupler and the power divider. Among these components, PLL determines the phase noise to a large extent. Usually the design of PLL is used with PLL chips integrated VCO. That makes the design simpler, but the phase noise is not better than that with PLL chips disintegrated VCO. Measures such as using fourth-order passive filter circuit and reducing the power supply ripple are adopted to optimize phase noise in addition. The test shows that the phase noise of the local oscillator is $-94\text{dBc}/10\text{KHz}$. The local oscillator must have the function that rapidly shut off in case an accident. The PIN switch plays the rule and it can shut off the oscillator in $1\mu\text{s}$. Due to requirements of the LHCD system, the local oscillator has three output ports, two ports must more than 30dBm , and another one must more than 20dBm . Before the production of the amplifier, coupler and the power divider circuit, the design must begin with theoretical calculation and regulated models built in the software ADS. In order to improve the stimulation accuracy, joint stimulation combined with the actual circuit is adopted in ADS. Test shows that the output power meet the requirements. Stimulations and experimental results fit well and the local oscillator has worked in LHCD system for months. That successfully indicate the reliability of the local oscillator.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 14 / 119

A Method for Diagnosis of Current in PF Magnet based on Inversion of Measured Magnetic Field

Author: Zichu Huang^{None}**Co-authors:** Shuxia Tian ; Zhenmao Chen

More and more Tokamak devices have been designed and constructed to challenge the controlled fusion problem, such as ITER, J-TEXT, EAST and HL-2M. The coil systems of the Tokamak devices are usually composed of Poloidal Field (PF) coils, Toroidal Field (TF) coils and Central Solenoid (CS) coils. Among them, the PF coils play a role for adjustment of the confining magnetic field to control the fusion plasma. In order to avoid the plasma disruption, it is necessary to know the correlation of the magnetic field in vacuum vessel (VV) with the current in coils and the plasma current and to predict the current distribution of PF coils from a distribution of perturbed magnetic field due to unstable plasma current. In this paper, taking the HL-2M device as an example, a method to predict the current distribution in the PF coils from the perturbed magnetic field in VV is proposed based on inverse analysis of the magnetic field information. Firstly, based on the Biot-Savart's law, a forward code to calculate magnetic field due to currents in PF coils, TF coils and the plasma current was developed. Secondly, an inversion code to predict current distribution of PF coils from the measured magnetic field was developed based on the conjugate gradient optimization method. The validity of the proposed inversion method and the corresponding numerical codes was investigated through reconstructing PF current distributions from several groups of perturbed magnetic fields with artificial noises.

Eligible for student paper award?:

No

M.OA2: Divertors and High Heat Flux Components / 120**Status of the ITER Cooling Water System Design****Author:** Giovanni Dell'Orco¹**Co-authors:** Ajith Kumar²; Andrea Ciampichetti¹; Biswanath Sarkar¹; Dinesh Gupta²; Donato Lioce¹; Fabien Berryuer¹; Jan Berry³; Mahesh Ashok²; Nicolas Ghirelli¹; Seokho Kim³; Steve Ployhar¹; Walter Van-Hove³¹ ITER IO² INDIA ITER³ US ITER**Corresponding Author:** giovanni.dellorco@iter.org

ITER Cooling Water System (CWS) is designed to reject all the heat generated in the plasma and transmitted to the In-Vessel components through the Tokamak Cooling Water System (TCWS) to the intermediate closed loop Component Cooling Water System (CCWS) and then to the environment via the open Heat Rejection System (HRS).

The TCWS is designed to remove the total peak heat load of about 1100 MW and is divided into three Primary Heat Transfer System (PHTS) loops, two Chemical and Volume Control System (CVCS) units, a Draining and Refilling system (DR) and a Drying System (DY). The TCWS has a safety role for the primary confinement of radioactive inventory due to Activated Corrosion Product (ACP) and Tritium content in the water. The three PHTS are: Vacuum Vessel (VV PHTS), Integrated Blanket ELMs and Divertor (IBED PHTS) and the Neutral Beam Injectors (NBI PHTS). The VV PHTS has also the safety function to provide the decay heat removal functions even when the other PHTSs are not available during off-normal accidental events like LOCA, LOSP etc.

The paper describes the main design challenges faced and the changes that have been carried out to prepare the CWS final design phase.

The paper also reports the main functional requirements for the CWS considering the phased installation, commissioning and operation of the CWS from the preoperational activity to the First Plasma and eventually to the nuclear DT phase.

Detailed information will be also provided about the physical and functional interfaces between CWS and the main clients (e.g. Vacuum Vessel, In-Vessel Components, Diagnostics, Power Supply, Cryoplant etc.) with the progress on the integration the CWS to the other systems in the Tokamak Complex as well as in the other non-nuclear buildings.

Eligible for student paper award?:

No

R.OP1: Diagnostics and Instrumentation II / 121**Joint plasma pressure diagnostic system of Beam Emission Spectroscopy and Ultrafast Charge eXchange Recombination Spectroscopy on EAST tokamak****Authors:** Yi Yu¹; Minyou Ye¹; Ran Chen²; Huajie Wang¹; Yinyin Li³; S. Zoletnik⁴; Bo Lyu²; Yuejiang Shi¹; Baonian Wan⁵¹ University of Science and Technology of China² Institute of Plasma Physics³ Institute of Plasma Physics⁴ KFKI-Research Institute for Particle and Nuclear Physics⁵ Institute of Plasma Physics, Chinese Academy of Sciences

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In this article we present the development of the joint plasma pressure diagnostic system on EAST tokamak, i.e., the Beam Emission Spectroscopy (BES) and the Ultrafast Charge eXchange Recombination Spectroscopy (UF-CXRS) basing on Neutral Beam Injection which are to diagnose two-dimensional plasma density and ion temperature simultaneously at the same spatial area with a time resolution at the order of $1\mu\text{s}$ and a spatial resolution of 1-3 centimeters. The main physical goal of this high-resolution joint plasma pressure diagnostic is to understand some underlying physics of turbulence, such as the formation of edge pedestal in L-H transition in which the plasma pressure gradient is the key parameter. These two diagnostics share the same light path. 128 BES channels have been constructed, and can measure plasma density distribution in a $20\text{cm}\times 10\text{cm}$ rectangular area in the cross section which is movable along minor radii ($\rho=0\sim 1$) by means of changing the angle of the mirror in the light path. Four of these 128 channels are chosen as test channels of UF-CXRS diagnostic to measure ion temperature. Components in the light path are carefully designed to rise to the great challenge of weak CVI emission light with a wavenumber of 529 nm, thinking of the high time resolution of microseconds. The time resolution raises three orders of magnitude than that of the traditional CXRS diagnostic on EAST tokamak. The whole design and some test results of this joint system are discussed in this article, together with first experimental results of BES part.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 44 / 122

Design and Analysis of the CFETR TF Coils with REBCO tapes

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Abstract—The China Fusion Engineering Test Reactor (CFETR), which has the potential to produce a fusion power above 2 GW, is being designed to bridge the gap between the ITER and Demo in China. The fusion power is strongly dependent on the plasma pressure to magnetic pressure ratio and toroidal magnetic field at the plasma major radius. The maximum magnetic field above 20 T is expected in the CFETR TF coil. The state-of-the-art low temperature superconducting (LTS) magnet has a relatively low coil current density in magnetic field above 15 T. Therefore, a high performance superconducting magnet with high temperature superconductor (HTS) is required to provide the superior superconducting property. The RE–Ba–Cu–O-coated (REBCO) tapes, which has the higher critical current density than LTS in magnetic fields above 15 T and endure a high tensile stress, will be used for the CFETR TF coil. The CFETR TF coil is composed of the REBCO HTS coils and Nb3Sn LTS coil.

This paper will describe the design of the CFETR TF coil. The electromagnetic and thermal-hydraulic analysis of the CFETR TF coil is performed.

Keywords—CFETR, Double-pancake (DP), REBCO coil, Superconducting magnet, Thermal-hydraulic behavior.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 86 / 123

COMPUTATIONAL MODELING AND CHARACTERIZATION OF PLASMA PULSED GUN FOR PLASMA WALL INTERACTION APPLICATIONS

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Previous research has demonstrated that linear plasma accelerators can achieve the energy, temperature and densities necessary to reproduce the conditions of plasma impact in a divertor plate [1]. Some initial experiments have been performed with plasma foci at University of Mexico [2]. At the Instituto Politecnico Nacional a new device, a flexible Plasma Gun that can be operated in snowplow and in deflagration mode is under development. The first step is to tailor the design in such a way that the parameters of the ejected plasma fall within the desired range. It is therefore of great importance to develop a computational model that can predict plasma parameters as a function of input parameters such as the electrical input circuit characteristics, the electrodes geometry, the operating gas pressure, the gas feed, the distance to the exposed sample, among others. Using a snowplow model such as the one developed originally by Hart as a starting point [3-4], a 2D-axisymmetric model for a coaxial plasma gun is constructed. This model includes circuit effects and calculates expansion of the plasma plume and its density and temperature, so reasonable mechanisms of the interaction of the sample with this plasma can be predicted and later be contrasted against experimental measurements once the physical device is constructed.

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Eligible for student paper award?:

Yes

R.OP2: PMI and Plasma Edge Physics / 124

Understanding tungsten divertor sourcing, transport and its impact on core impurity accumulation in DIII-D high performance discharges

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Using two toroidal rings of isotopically enriched tungsten-coated metal inserts in the predominantly carbon DIII-D divertor, detailed information on W sourcing, transport, and core contamination has been acquired over a range of high performance plasma conditions. W is the planned plasma facing material for the ITER divertor, but its contamination of the core plasma can have deleterious effects on fusion gain. To mitigate W core contamination in ITER and beyond, it is necessary to develop a comprehensive understanding of the W impurity pathway from the divertor into the core. Experiments were carried out on DIII-D featuring the first-ever use of distinct W isotope mixes in a fusion device, and were coupled to inductively coupled plasma mass spectrometry of removable midplane collector probes to determine the W leakage from localized regions in the divertor into the main chamber.

The evolution of W-C mixed-material surface layers in the divertor, formed via continual erosion and re-deposition of W and C ions, was also quantified via removable C inserts mounted on a divertor collector probe exposed during ~ 100 s of repeated L-mode discharges. Contrary to predictions from preliminary ERO modeling, W coverage was observed to be nearly constant with both major radius and exposure time on the inboard half of the probe near the outer strike point (OSP), while being much larger and increasing with exposure time at the outboard side in the far scrape-off-layer (SOL). Depth profiles from Rutherford backscattering spectrometry reveal thick mixed-material deposits on the outer half of the probe inserts, leading to higher than expected W retention in the redeposited C layers.

Edge-Localized-Mode (ELM)-resolved spectroscopic measurements of W sourcing conducted during H-mode discharges with high spatial resolution indicate, for the first time, that large ELMs shift the peak W erosion rate radially outwards, in addition to the broadening of the erosion profile because the ELM wetted area increases with ELM size similar to observations in the JET-C divertor [1]. Detailed analysis of intra-ELM W sourcing shows that C impurities and D fuel ions contribute equally to W divertor sourcing during ELMs, in contrast to JET-ITER-like Wall (ILW) where D ions were the main contributor [2].

High performance discharges ($P_{AUX}=14$ MW, $H_{98}=1.6$, $\beta_N=3.7$) with high ELM frequency ($f_{ELM}\sim 200$ Hz) run during this campaign show roughly equal performance with similar scenarios run previously with the all-C divertor. These discharges exhibit low core W concentrations (few 10^{-5}) presumably due to flattening of the main plasma density profile via on-axis electron cyclotron heating (ECH), in line with the experience on other devices with high-Z plasma facing components [3]. In these discharges, W impurities are predominantly transported to the midplane from the OSP region rather than the far-SOL region. Conversely, in scenarios when f_{ELM} drops to 60 Hz, the W impurities are shown to transport equally from the OSP and far-SOL regions.

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*Supported by US DOE under DE-FC02-04ER54698, DE-AC02-09CH11466, DE-AC05-00OR22725, DE-AC52-07NA27344, DE-FG02-07ER54917, DE-AC04-94AL85000.

Eligible for student paper award?:

No

T.OA1: Diagnostics and Instrumentation I / 125

Design, Manufacturing, and Integrated Testing of the ITER Port Instrumentation

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At ITER more than 50 different diagnostics are under development for the tokamak. The diagnostic systems are designed to be integrated within the interspace and port cell support structures of 27 upper, equatorial and lower ports. Basic instrumentation and control (I&C) is required to monitor the temperatures at selected locations of the port plug and interface support structure and for the electric heaters used for baking of windows and thermal stress compensation. Spare measurement channels have to be provided for future use.

Currently the ITER project is transitioning from the detailed design phase to manufacturing, testing and integration in preparation for integrated commissioning. The focus of the work is on the first plasma diagnostics for which system integration in the equatorial ports 11 and 12 is essential. Since the port system I&C is required for many port systems, the development is already quite advanced with manufacturing and acceptance testing currently taking place. Furthermore the port system I&C is typical for industrial plant I&C and can therefore serve as an example for those plant systems.

The design process starts with the requirement capture from all relevant sources, continues with a description of use cases and operating procedures, and is followed by the functional analysis including the definition of all the variables providing the interface with CODAC through its networks. The software implementation process is based on the CODAC Core System (CCS) and CODAC provided tools. The integrated testing follows a set of test campaigns starting from acceptance of the installed hardware and of the source code in the software repositories.

This paper presents the development process of the port system I&C through all lifecycle phases from design to site acceptance and summarizes the test results.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 35 / 126

Recent progress of pellet injection system in Experimental Advanced Superconducting Tokamak

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Pellet injection, which is regarded as the most promising technique for the research of edge plasma physics, can control the edge localized mode (ELM) [1] and reduce the power threshold of L-H transition [2]. Since the installation of the 10 Hz pellet injection system [3] in Experimental Advanced Superconducting Tokamak (EAST), lots of experiments have been carried out. The 10 Hz pellet injection system in EAST can continually produce pellets with both diameter and length of 2 mm, containing $\sim 3.78 \times 10^{20}$ atoms in each pellet. Except for the normal fueling effect, high-confinement (H-mode) plasma was achieved by injecting frozen deuterium pellets in EAST. Interesting phenomena of simple and two-stage low-high confinement (L-H) transitions are observed in EAST with radio frequency heating after shallow pellet injection. The results of the L-H transitions induced by pellets are discussed in detail with different theories. It is found that pellet injection in EAST can reduce the power threshold of H-mode. Furthermore, the pellet-induced edge density gradient is one of the important parameters affecting the L-H transition. Comprehensive researches will be carried out in the next campaign with the development of a new 50 Hz pellet injection system recently [4, 5] in EAST, which is capable of injecting pellets with different sizes. Besides, it is also observed in EAST that a deep penetration pellet can cause severe snake-like perturbation in the core plasma region [6]. This snake phenomenon, which was clearly monitored by the soft X-ray diagnostic, had a long life time of ~ 1 s. These investigations prove that pellet is a powerful tool to investigate not only the edge plasma physics but also the core plasma physics. This research is funded by the National Nature Science Foundation of China under Contracts No. 11625524, No. 11321092, and No. 11605246 and the National Magnetic Confinement Fusion Science Program under Contract No. 2013GB114004.

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Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 107 / 127

The Disturbance Analysis for Ultrasonic Doppler Profile Measurements Through Numerical Simulation

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The liquid PbLi and PbBi alloys have been selected as one of the most promising coolant materials in fusion blanket and accelerator driven subcritical (ADS) reactor, respectively. The velocity distribution in the flow channel of blanket will greatly affect the efficiency of heat transfer and materials corrosion, so it's necessary to carry out relevant researches to obtain the detailed flow field. Ultrasonic Doppler velocimetry (UDV) is considered as one of the best methods measuring opacity fluid. In the process of UDV measurement, the measuring accuracy is one of the most important problems and large bias is observed in measuring results. The reason for the bias is that there are three factors, intersection between measuring volume and wall, refraction of the ultrasonic, ringing effects of piezoelectric element, which mainly influence the measuring accuracy. However, it's well known that the ultrasonic is a kind of mechanical wave, which will lead to vibration of the medium in the process of propagation. The vibration of medium will cause disturbance to flow field and then lead to measuring bias, so it should be also considered as a key factor in flow field measurement. However, the influence of vibration is usually ignored and no researches were performed on it. Hence, it's necessary to research about it.

In this paper, ANSYS analysis is adopted to investigate the effect of vibration of medium. The vibration of medium leads to the acoustic pressure variation of fluid which can deduce the vibration velocity. The acoustic pressure in flowing fluid is calculated through numerical simulation and the two

factors, excitation voltages and backing layer, are analyzed. The results show that a large disturbance velocity up to 10mm/s is generated in heavy liquid metal and it's proportional to excitation voltages. The disturbance is slowly growing up with the thickness of backing layer increasing from 2mm to 6mm, but no obvious variation is obtained when the thickness of backing layer is larger than 6mm. So, it's very meaningful for bias analysis in flow field measurement based on the results. Besides, it can support the design of UDV sensor and conduct the parameters setting in experiments.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 6 / 128

The Design of DRAGON-V Loop for Key Technique Verification of Liquid PBLI Blanket

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The liquid Lead Lithium (PbLi) blanket is one of the most promising blanket concepts for fusion reactors. Aiming to better develop the PbLi blanket technology and realize engineering application, key issues of PbLi blanket should be investigated such as material corrosion, the magnetohydrodynamic (MHD) effect and so on. In addition, the integrated tests and engineering design validation of PbLi blanket module should be performed as well. So, it is necessary to develop experimental platforms to study the key issues of PbLi technology for fusion reactor.

At present, a series of PbLi experimental loops have been designed and built successfully by FDS Team such as DRAGON-I/II and DRAGON-IV. Some experiments have been conducted to investigate the corrosion behaviors of CLAM steel in magnetic field, the purification technology of liquid PbLi and MHD pressure drop test. To support the engineering design validation of DEMO blanket with the parameters covering the requirements of ITER-TBM and China DEMO, a dual coolant thermal hydraulic integrated experimental Loop DRAGON-V was designed, which can be used to study the integrated experiments under the multi physical field conditions for fusion reactor. It is composed of lead-lithium loop and helium loop. The maximum temperature in the test section is designed to be 1100 °C, the maximum flow rate of PbLi can reach 40 kg/s, and the magnetic fields is up to 5 T. The maximum helium pressure is 10.5 MPa. It can carry out the research of material corrosion under different magnetic fields, MHD test for components of liquid blanket and LOCA. The obtained findings can support the development of the key techniques in-pile and the engineering design of China DEMO reactor. Besides, it can also be used for the advanced Generation-IV reactors and civil application.

Eligible for student paper award?:

No

W.OA1: Materials II / 129

Effect of Heat Treatment on Anisotropic Tensile Behavior of CLAM Steel Fabricated by Additive Manufacturing

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Fusion reactor blanket needs to withstand the 14.06 MeV high energy neutron irradiation, high heat flux, high nuclear thermal deposition and complex electromagnetic and mechanical loading and so on, which put forward a high requirement for the structural materials and the quality of components manufacturing. Reduced-activation ferritic/martensitic (RAFM) steel has been developed as the structure materials for future fusion reactors due to its excellent thermophysical and mechanical properties. China low activation martensitic (CLAM) steel, a kind of RAFM steel, has been chosen as the primary structure material of China ITER TBM and China fusion engineering test reactor (CFETR) blanket. In the harsh service environment of fusion reactor, the blanket is generally designed with high density and complex embedded cooling channels, which is a challenge to the traditional manufacturing technology.

Additive manufacturing (also named 3D printing) is an advanced precision manufacturing technology, which can be used to manufacture the parts that are difficult for traditional methods. It has the advantages of near net forming, high forming efficiency of complex structural components and good integrated. It is a new exploration to apply the additive manufacturing technology to the complex structure manufacturing of fusion reactor blanket, which is of great significance to the development of the preparation technology of the components with complex structures. The mechanical properties of the parts, which were fabricated layer by layer with additive manufacturing, are anisotropic in the directions of the parallel layers direction and the vertical layers direction. Therefore, it is important to research the anisotropic mechanical behavior of CLAM steel and to explore the technique. In this study, the selective laser melting (SLM) was adopted for the additive manufacturing of CLAM steel. The hot isostatic pressing (HIP) and quenching and tempering heat treatments (HT) were performed. The effect of HT on microstructures and tensile properties at different status were investigated. The results showed that tensile strength at parallel layers direction and the vertical layers directions were 966 MPa and 880 MPa at SLM status. After the HIP quenching and tempering HT, the tensile strength were reached to 694MPa and 684MPa, and then the tensile strength of two directions were so closed due to the recrystallization of grain during the HT conditions. The results of this research were of great scientific research significance and engineering application value for the techniques application in the key components manufacturing of the fusion reactor blanket.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 36 / 130

Application of PAUT in CFETR vacuum vessel austenitic stainless steel welding R&D

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Full penetration welding and 100% volumetric examination are required for all welds of pressure retaining parts of the CFETR(China Fusion Engineering Test Reactor) Vacuum Vessel (VV) according to the design manual. But not every welding joint can be tested with RT due to the structure and welding position. Therefore the ultrasonic testing (UT) has been selected as an alternative method. Considering the misjudgment and undetectable in the austenitic stainless steel welding by the traditional ultrasonic testing method, this paper introduce the application of PAUT(phase-array ultrasonic technique) in the CFETR VV R&D. Base on the ultrasonic simulation and dynamic focus, the precision of the defect position and the signal/noise(S/N) can be improved. The PAUT show excellent detectability and applicability for the austenitic stainless steel weld in the CFETR VV.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 72 / 131

Web Services for 3D MHD Equilibrium Data at Wendelstein 7-X

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Wendelstein 7-X (W7-X), the first fully-optimized stellarator experiment, started its operation in December 2015. W7-X research aims for good plasma confinement and the demonstration of steady-state operation. This could make the stellarator a serious option for a future fusion power plant. Magnetohydrodynamic (MHD) equilibrium data is needed at W7-X for data analysis and plasma operation (e.g. on-/off-axis heating scenarios). For axisymmetric tokamak configurations, equilibrium reconstruction in 2D can be done in nearly real-time. Due to the geometry of stellarators a three-dimensional MHD equilibrium model is needed. This leads to codes which are much more complex and time consuming. For example the VMEC code [1], a widely-used 3D code which assumes nested flux surfaces, takes minutes to hours for such computations. Therefore the VMEC equilibrium data for W7-X configurations has to be calculated in advance and stored in a database.

Web service technology is used to provide convenient access to equilibrium calculations by the whole W7-X team. The usage of standard web protocols allows the integration in client software written in almost any programming language. The VMEC web service provides access to equilibrium calculations via SOAP interface

and REST API. The user can execute VMEC calculations and receive the results of completed runs. A website provides documentation for all functions as well as a listing of all stored configurations and previews of configuration parameters and interactive flux surface plots. The VMEC web service also provides access to the

W7-X Reference Equilibria: a managed collection of VMEC configurations for W7-X experiments. This includes calculations for the standard magnetic configurations of Wendelstein 7-X and also the limiter configuration, which was used in the first operational phase of W7-X.

At Wendelstein 7-X a growing number of web services are implemented and in use. For example, the W7-X Archive Web API provides unified access to the data of all diagnostics and also machine operation data. Further examples are web services for Biot-Savart calculations, magnetic field line tracing, and function parameterization for W7-X configurations, as well as web service interfaces for a coil geometry database and a database for 3D meshes of machine parts and components. This allows service orchestration: results can be used directly as input for the VMEC web service and similarly the VMEC equilibrium data can be used as input for other web services. In this way the web service approach leads to a better interoperability of codes and increases the traceability of calculation results. Most of the services use generic data models and can therefore be used for calculations for other machines as well. This contribution presents the current state of the equilibrium web services at Wendelstein 7-X.

[1] Hirshman, S., van Rij, W. and Merkel, P. "Three-dimensional free boundary calculations using a spectral green's function method", *Comput. Phys. Comm.*, 1986, 43, 143-155

Eligible for student paper award?:

No

R.OP5: Experimental Devices II / 132**Tokamak design and maintenance scheme trade off Application on CFETR****Author:** Eric Villedieu¹**Co-authors:** Yuntao Song²; Cheng Yong³; Juncheng Zhan³; Yang Li³; Christian Dechelle⁴¹ CEA² Institute of Plasma Physics, CAS³ Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China⁴ CEA-IRFM, 13108 Saint Paul lez Durance, France**Corresponding Authors:** liyang@ipp.ac.cn, chengyong@ipp.ac.cn, eric.villedieu@cea.fr, zhanjc@ipp.ac.cn, christian.dechelle@cea.fr, songyt@ipp.ac.cn

The new tokamak generation will be characterized by the necessity of full remote maintenance for most of the critical components. Because no human intervention can be envisaged, the remote systems will have to prove high reliability and rescue capabilities. The availability of the fusion facility will have to be maximized, as a consequence, the efficiency of the maintenance equipment will become a key factor. To fulfill these constraints, the remote maintenance principles will need to be simplified as much as possible and be taken into account at the early stages of the tokamak design. This is an iterative process between Tokamak design and Maintenance system design in order to reach the best possible trade off.

Such an exercise of this iterative work was done on the CFETR preliminary design.

Main targets were established in the view of optimization of the remote maintenance, then an assessment of the first maintenance scenario, envisaged for the in-vessel components, was done and alternative solutions were proposed. This process was repeated during meetings with the tokamak sub-system owners.

The paper will present the results of a first phase of optimization. This work has provided guidelines for the main tokamak sub-systems like the magnet configuration, the blanket and divertor modularity, the neutron shielding and cryostat arrangement, the transfer of components and the hot cell configuration.

It also produced some innovative solutions for the overall maintenance scheme like the implementation of a hot cell at the top of the tokamak hall avoiding cask transfers. On this basis a CFETR maintenance scenario is proposed.

The paper will address the following topics:

- Preliminary design and optimization drivers
- Blanket and divertor design proposal
- Blanket and divertor maintenance system
- Equatorial plug maintenance system
- Cask, transfer and double door system proposal
- In-cryostat maintenance principle
- Pending issues to be analysed

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 75 / 133**Numerical–experimental benchmarking of a probabilistic code for prediction of Voltage Holding in High Vacuum, for ITER N-NBI.****Authors:** Nicola Pilan¹; Atsushi Kojima²; Ryo Nishikiori³; Masahiro Ichikawa²; Junichi Hiratsuka²; Specogna Ruben⁴; Antonio De Lorenzi^{None}; paolo bettini⁵; Mieko Kashiwagi³

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In the framework of the program for the construction of 1 MeV–16 MW negative neutral beam injector (NNBI) for ITER, an R&D activity on voltage holding in vacuum has been initiated since 2009, aimed at supporting the design, construction, and development of the NNBI accelerator. For this purpose the voltage holding prediction model (VHPM) previously developed [1] has been updated. In the VHPM the effect of the electric field anode and the electric field cathode on the probability of breakdown is evaluated, by two exponents: alpha and gamma. On the basis of the experimental results from different test stands and of detailed 3D numerical simulation of the corresponding electric field configurations, the predictions of the VHPM numerical code have been benchmarked. New exponents, alpha and gamma have been proposed to obtain a more precise location of the weak point of the system and a better prediction of the maximum withstanding dc voltage in high vacuum.

Ref.

[1] N. Pilan, P. Veltri and A. De Lorenzi, IEEE Transactions on Dielectrics and Electrical Insulation Vol. 18, No. 2; April 2011

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 76 / 134

Control and protection system for the W7-X ECRH plant –experience from the first and plans for the next campaign

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W7-X is a steady state capable optimized stellarator. The main heating system is electron cyclotron resonance heating (ECRH) operating at 140GHz providing up to 9MW microwave power.

A set of diagnostics has been developed to protect the machine from non absorbed ECRH power which can easily damage in vessel components.

The power is launched into the machine by front steerable quasi-optical launchers in X- or O-mode. While in X-mode the first pass absorption is ~99%, it is only 40... 70% in O-mode. The non absorbed power hitting the inner wall is measured by waveguides embedded in the first wall (ECA diagnostic). In order to prevent the inner wall from overheating or arcing, a near-infra red sensitive video diagnostic with a dynamic range of 450...1200°C was integrated in the ECRH launchers. Thermal calculations for the carbon tiles predict a temperature increase above the detection threshold for scenarios of plasma start-up failure or poor absorption on a time scale of ~100ms and the risk of overheating after ~300ms. However, no temperature rise above the detection threshold could be observed in experiments with failed break down, i.e. poor ECRH absorption for up to 100ms.

The stray radiation level inside the machine is measured by so called sniffer probes which were designed to collect all radiation approaching the probing surface independent of incident angle and

polarization. Five sniffer probes are installed at different toroidal positions. They were absolutely calibrated.

The sniffer probes are integrated in the ECRH interlock system. During the first operational phase of W7-X this was the only available plasma interlock system. The signal quality proved to be high enough for a reliable termination in case of poor absorption. After a breakdown phase of ~10ms, the sniffer probe signals dropped by more than an order of magnitude. However, especially in the very first days of operation, most discharges died by a radiative collapse due to impurity influx. In this case the heating power was reliably switched off due to the increased level of stray radiation.

During OP1.1 the gyrotrons which are mostly capable of delivering >900kW power were operated at reduced power to increase the reliability. At maximum power there is an increased risk of losing the main mode in the gyrotron and thus a stop of RF emission. This causes an interlock for the gyrotron. An intelligent control system being able to operate stable at maximum output power is currently being developed. One approach is to bring the gyrotron back in operation after a mode loss by switching off the HV supply for a short time (<1ms) and then back on at a slightly lower value. The second approach is a feedback control system to stabilize the output power by identifying a precursor signal for a mode loss and minimizing it. The mode activity at parasitic frequencies has been identified a possible precursor for a feedback control system.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 62 / 135

Three confinement systems - Spherical Tokamak, Advanced Tokamak and Stellarator: A comparison of key component cost elements

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Since the 1950's Next Step fusion devices and power plant studies have been developed for a number of magnetic confinement systems but an open question remains...can a magnetic fusion device be simplified to the point where it will be cost competitive and operate with high availability? Concept designs based on the advanced tokamak (AT), spherical tokamak (ST) and the quasi-axisymmetric stellarator (QAS) option have progressed in recent years through a series of PPPL studies with an underlying intent to improve the engineering feasibility of each, giving special attention to concepts that simplify the device configuration and improve maintenance features. For the spherical tokamak option, design details centered on a 3m Fusion Nuclear Science Facility concept that evolved to incorporate vertical maintenance, HTS magnets, a small inboard DCLL blanket and a liquid metal divertor. In collaboration with the K-DEMO and CFETR concept study teams the AT design has evolved to increase plasma component access within a vertical maintenance approach using enlarged TF coils incorporating a low and high field Nb3Sn winding pack that can provide a peak field of 16T. A recent PPPL stellarator study focused on simplifying the stellarator winding topology to improve access to in-vessel components; combining coil optimization with winding surfaces that incorporated geometry constraints specified by engineering. This study centered on a 1000 MW power plant design with a tokamak like vertical maintenance scheme that allows access to remove large segmented internal blanket sectors.

Results of these three confinement studies will be presented to highlight concepts that simplify each device configuration and improved their maintenance features. Scaling each option to a common 1000 MW net electric power plant mission allows comparisons to be made of key cost elements such as to major core component sizes, sizing of the test cell or external facilities needed for on-site construction or facilities to handle and store activated in-vessel components.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 15 / 136

Mechanical Designs for High Magnetic Field Tests for ITER Applications

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In the ITER (International Thermonuclear Experimental Reactor) program, core imaging X-ray spectrometer (CIXS) and electron cyclotron emission (ECE) are two diagnostic systems in the US ITER project, where X-ray Dectric PILATUS single-photon-counting pixel detectors for X-ray energy and piezo actuators are required to operate under the conditions of high magnetic fields and the accompanying rapid field transient rates. The two devices were tested under the conditions in Princeton Plasma Physics Laboratory. For the tests, a transrex power supply was employed to provide high intensity of magnetic fields (up to 3T), and a series of mechanical devices were designed and made to carry and secure the testing devices in the magnetic field. In addition to structural integrity, material magnetism was a major concern and an analysis of the magnetic properties was carried out. In this paper the testing fixtures are described, mechanical setups, instrumentation, generation of the high magnetic field, safety aspects and procedure are also introduced. With the testing setups, the tests for CIXS and ECE components were successfully completed, and the results are provided to ITER applications.

Eligible for student paper award?:

No

R.OP5: Experimental Devices II / 137

CODAC Core System for ITER plant system I&C

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Construction of the ITER project is distributed among its many members. The ITER plant systems are built including their local controls by many partners all around the world, to be delivered as in-kind contributions to the ITER site after factory acceptance testing. A total of 171 separate Instrumentation and Control (I&C) sub-systems, for power supplies, cooling water, cryogenics, magnets, fuelling, vacuum, vacuum vessel and cryostat, heating systems, diagnostics, etc. are currently being designed and manufactured for ITER, covered by 101 procurement arrangements. After delivery, these systems will be tested, integrated and maintained by the ITER central team for the commissioning and operation of the tokamak device.

To cope with this level of work distribution and the resulting integration risks, the ITER Control System Division has developed and maintained standards for ITER I&C, defined in technical specifications and supported by services, covering hardware procurement as well as software design and development.

One of these services is to provide the suppliers with a dedicated software distribution, named CODAC Core System (CCS), comprised of frameworks, tools and components for developing the plant system I&C according to the ITER defined standards. This software distribution is based on Red-Hat Enterprise Linux as operating system and EPICS as control system framework. Onto this, it adds control software either developed by the ITER Organization or specifically adapted for our requirements from existing solutions. CODAC Core System has been released at least twice per year since 2010, with increased functionality, stability, and completeness. In parallel, the quality control mechanisms have been developed that allow reaching the required integrity levels, and the support organization has been adapted to cope with the increasing user assistance requirements.

This paper describes the technical and organizational solutions adopted for enhancing homogeneity in the ITER Instrumentation and Control (I&C) software and improving control over its development and tests. It also reports observed results in the current development and test activities and presents the roadmap towards commissioning and operation.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 57 / 138

A Method to Alleviate the Long History Problem Encountered in Monte Carlo Simulations via Weight Window Variance Reduction

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Implementing weight window (WW) is a usual method for variance reduction (VR) of Monte Carlo simulation, however as for a complex and large model simulation it frequently encounter the long histories (LH in abbreviation) problem in parallel computing. LH behavior shows as the running time of a single particle history is significantly longer than that of normal histories. It would take a disproportionate amount of time for Monte Carlo simulation to accomplish and place a detrimental effect on the efficiency of parallel computing. In this paper, the investigation of reason that causing LH was carried out firstly. A simple dog-log model was constructed to observe and analyze the LH phenomenon. Then comparative tests were carried out on a 3D model of the Chinese Fusion Engineering Testing Reactor (CFETR) with three approaches these are: a) analog running without any VR techniques; b) normal weight window VR technique; c) a novel approach proposed in this paper of limitation of weight window splitting. The results show that a suitable set of parameters in the improved WW module significantly improves the efficiency of variance reduction performance in parallel calculation, making the long history problem tractable without biasing results.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 26 / 139

Physics and Geometry Design of Lower Divertor Upgrade in EAST

Tokamak

Authors: Houchang Xu¹; Zibo Zhou²; Lei Cao³

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Abstract-Experimental Advanced Superconducting Tokamak(EAST) device is a D-shaped full superconducting tokamak with actively water cooled plasma facing components. Before this upgrade, three generations of divertors, which, respectively, are steel divertor and carbon divertor and International Thermonuclear Experimental Reactor (ITER)-like divertor have designed. To achieve long pulse and high β H-mode plasma, new plasma position and shape are calculated and optimized in 2016 for EAST. The new geometry of lower divertor heavily relies on numerical simulations of the plasma in EAST.

New divertors are designing to fit the high β H-mode plasma and endure the heat flux up to 10 MW/m². To solve this issue, the lower carbon divertor will be upgrading in the future in EAST, which is in conceptual design phase. In consideration of the structure profile and function in EAST tokamak, in conceptual design phase these questions will be solved as follow. Firstly, the divertor should be better designed with advanced physical operation mode. Secondly, the divertor should be advanced geometry and high efficient cooling structure. The cooling circuit and the support systems of the component are installed on the vacuum vessel. The size of the space under the divertor should be consideration, which is very important. Thirdly, the cooling structure and maintenance of the divertor are also introduced in the paper.

In consideration of physics and geometry design of lower divertor upgrade in EAST Tokamak, in the paper, mainly introduce the research progress of the fourth generation of divertors in EAST, much effort was focus on the divertor configuration and geometry.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 37 / 140

Pumping Performance Calculation of HL-2M in-vessel Cryopump based on Monte Carlo method

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Sufficient pumping speed and good pumping performance should be guaranteed for the HL-2M Tokamak. Technical parameters were obtained by directly simulating with the Monte Carlo method for HL-2M in-vessel cryopump under molecular flow conditions. The predicted pumping speed of bare pump is 51.29 m³/s for H₂, 38.04 m³/s for D₂ and 24.94 m³/s for He. An advanced divertor system will be installed, located on the floor of the HL-2M vessel. Its pumping speed contained conductance of divertor is 20.11 m³/s for H₂, 15.14 m³/s for D₂ and 12.50 m³/s for He. It shows

that the pumping effectiveness of the cryopump be affected by the structure of divertor greatly. The pumping speed of the cryopump can be influenced by the sticking coefficient, results be obtained by analyzing different sticking coefficient (vary ± 0.05). The numerical and deduction results show that the sticking coefficient has small effect on pumping speed for H₂ and D₂, but obviously affects the pumping speed for He. The pumping process dynamic evaluation results show that the cryopump has a quick time response.

Eligible for student paper award?:

No

R.OP4: Stellarators / 141

Mechanical Monitoring Issues in Preparation to Next Step of W7-X Operation

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The largest modular stellarator Wendelstein 7-X (W7-X) has successfully passed first phase of operation in Greifswald, Germany. The sophisticated W7-X superconducting magnet system with its non-linear support system has been carefully monitored using an extensive set of mechanical and temperature sensors. The paper focuses on detail consideration of cyclic magnet system behavior during operation with limiter configurations of plasma. Measurement results are carefully compared with predictions from updated numerical models and critical issues are highlighted. As a result, the structural monitoring tool is extended to follow enhanced requirements and expectations.

The work is a preparation for upcoming more demanding phases with longer plasma pulses to guarantee safe and reliable W7-X operation with different divertor and scraper element configurations. The procedure to establish required sensor configurations, to analyze and to release new plasma regimes being compatible with W7-X component design values is also described.

Eligible for student paper award?:

No

T.OP2: Fueling, Exhaust, and Vacuum Systems / 142

Refined Multiphysics Analysis of W7-X Cryopumps

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Ten identical cryopumps (CVP) are to be installed in corresponding divertor volumes of Wendelstein 7-X (W7-X) stellarator before commencing the steady state phase of operation (OP 2). Each CVP is typically made of two units connected by a transfer line and is fed with a dedicated plug-in. The units consist of water baffle, liquid nitrogen (LN2) baffle, helium panel and LN2 cooled housing. All components are expected to be well cooled with the available cooling capacity during long pulse plasma operation in order to maintain the helium panel at about 4 K and hence ensure the desired absorption rate. The LN2 cooled housing has to minimize both the effect of electron-cyclotron resonance heating (ECRH) and the thermal radiation from backside of in-vessel components on the CVP 80 K shield and 4 K elements. Moreover, the ECRH is unevenly distributed in W7-X which requires analyses of several cases and sophisticated cooling scheme as described. The paper presents thermal behavior of CVP components and is followed by the discussion of several important issues for the assessment. In addition, eddy current and electromagnetic (EM) forces on CVP copper components are analyzed for the events of fast discharge of main superconducting coils, plasma current decay, alternative current in control coils and fast discharge of trim coils. Moreover, sharing of eddy currents between plasma vessel shell and attached CVP is estimated. Structural analysis taking temperature gradients and EM forces into account finalizes the study with some conclusions and recommendations.

Eligible for student paper award?:

No

M.OP1: Plasma Operation and Control / 143

A first analysis of JET plasma profile based indicators for disruption prediction and avoidance.

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Disruptive events still pose a serious problem for the protection of in-vessel components of large size tokamak devices, representing therefore a key aspect to be considered for the design of next step fusion devices such as ITER and DEMO. If an efficient mitigation is strongly required to avoid damage and preserve the structural integrity of the machine, efficient avoidance schemes are needed to possibly bring the plasma back to a safe operating condition. In this framework, disruption prediction plays a key role and in the last few years a substantial effort has been devoted to developing more sophisticated prediction systems and improving their performance both in terms of success rate and warning time. Many of the presently developed disruption predictors mainly rely on MHD markers related to still rotating modes and, especially, to locked modes, which are basically the final precursor of most of the disruptions. Nevertheless, in many cases the detection warning time is still unsatisfactory with respect to avoidance requirements, and a significant step forward needs to be taken.

This work deals with the development of “plasma profile based indicators” for disruption prediction and avoidance in JET, where parameterized peaking factors have been implemented for electron temperature, density and plasma radiation profiles. The basic interplay of the time evolution of different profiles will be described in relation to the phenomenology characterizing specific disruption types together with the relevant time scales. Furthermore, a statistical analysis aiming to describe differences and boundaries between the safe and the disruptive space as well as among specific types of disruptions will be presented, discussing the implications in terms of disruption prediction and avoidance.

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

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 77 / 144

Advances in Technology, Performance, and Power and Polarization Measurements for the ECH System on DIII-D

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The DIII-D electron cyclotron heating system (ECH) has six gyrotrons installed at this time and is operated for injection into the plasma of rf power up to 3.6 MW at 110 GHz frequency. The rf power injected at the tokamak is measured on a shot to shot basis with a calibration based on the heat deposition in the gyrotron water cooling circuits.

A technique for calibrated ECH power measurement using both orthogonal polarizations of the transmitted rf wave at the last miter bend in the line was tested. Polarization scans for each system show H-plane and E-plane rf waveforms can be combined using square law detectors to provide a reliable calibrated power signal at the closest access point near the tokamak. Previous attempts to calibrate the power at this location were limited by the detection of only one polarization component at the last miter bend.

The elliptical polarization of the injected rf wave was measured for all the transmission lines. Final wave polarization is controlled using a pair of corrugated mirrors installed in miter bends. This allows for the launching the extraordinary mode that is absorbed at the second resonance for different plasma configurations and injection angles for heating or current drive. Two types of corrugated mirrors with different power handling capability were investigated. The smaller size mirrors performed better in terms of ellipticity control than the larger size mirrors with tapered mode convertor.

System upgrades include a new operating frequency, a new upgraded collector map measurement system better adapted to different configurations of collector RTDs, 4-port power monitors for reflected power and polarization measurements, and robust beam refraction protection using a density interlock, visible cameras, visible light monitors, and reflected power monitors used as sniffers. Increased levels of reflected power can indicate low absorption in the plasma or arcing in the launcher. Updates of the protection circuits to allow recovery after faults such as rf dropouts are included in the future plans for the system.

A newly designed depressed collector gyrotron in the 1.5 MW class, operating at 117.5 GHz, will be added to the ECH system. This new gyrotron has achieved 1.8 MW for short pulses during factory testing and is expected to be installed and operated in the spring of 2017. The system expansion is expected to reach a total installed power of over 11 MW and a total injected power of 8 MW with ten gyrotrons, some operating at 117.5 GHz and some at 110 GHz.

This material is based upon work supported by the U.S. Department of Energy, Office of Science,

Office of Fusion Energy Sciences, using the DIII-D National Fusion Facility, a DOE Office of Science user facility, under Award DE FC02-04ER546981.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 100 / 147

Measurement system of PSM HVPS for neutral beam injection on HL-2A

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Co-authors: weibin Li ¹; qinghua Ren ¹; xiaohui Mao ¹; qing Li ¹

¹ SWIP

In order to improve the experimental parameters on the HL-2A device and meet the requirements of the HL-2A modification, the capability and pulse duration of the auxiliary heating system should be improved greatly. The high voltage power supply (HVPS) which is based on PSM technology is a method of controlling the total output voltage of many identical DC choppers connected in series by means of managing the on and off of modules step by step and modulating their pulse widths in a certain sequence. The neutral beam injection auxiliary heating system should have high output power (80kV/200A), high accuracy of the output voltage (1%), flexible control, very high overall efficiency, very low amount of stored energy (when loads are arcing) and can be switched off immediately (< 15us). It is necessary to design stable and precise measurement system that is basic premise and important guarantee for these features, especially for the sampling data, system control and protection reliability. The paper describes the measurement system (the slow signal bandwidth is about 0-20kHz, the fast signal bandwidth is about 0-250kHz) which is composed of various kinds of transducers (HV divider, AC and DC current transducer), isolation transmission, industry control machine and PCI based data boards. The experiment results show that the measurement system is flexible and stable.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 101 / 148

Design and implement of Varying Frequency Three-phase Synchronous Signal processing system Based on modern signal processing

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In HL-2M the magnetic field power supply includes CS power supply and sixteen poloidal field power supplies. Each power supply is consists of three-phase full bridge thyristor converters and phase control is used to fire thyristor. The AC power of magnetic field power supply is provided by a

six-phase motor generator with two Y windings of shifting 30° . The AC voltage waveform is distorted due to heavy loads, and the frequency of generator outputs is changed during the pulse of plasma shot. So it is difficult to obtain clean and precise synchronous voltage for thyristor firing system. In order to enhance control precision and reliability of power supply, a new three-phase synchronous signal process test platform is developed. The simulation results in test platform show that the method is feasible. Then, the new synchronous processing system is founded. Through real-time data acquisition system three-phase synchronous AC signal are inputted. Digital filter technology is used to deal with input signal and according real-time frequency and phase bias compensation phase is realized by FPGA. And different phase of synchronous signal can be gained through this new system. The experimental results show that the three-phase synchronous signal processing system meets the design requirement.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 34 / 149

Numerical simulation of particle dynamics in the magnetic mirror

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The motion of electron in the magnetic field from a pair of current-carrying circular coil was studied using the standard numerical method for solving the differential equations. The situations of two coils with the same and reverse directions current were studied respectively, and the results were compared.

In both of the two magnetic fields, the trajectory of an electron (or other charged particle) is sensitive to the initial state. Both types of magnetic mirrors are not capable of effective magnetic confinement for various initial states of electrons, even if confined at the beginning, but in subsequent movements, the electrons may escape at any time (when the time is long enough after).

【Key Words】 magnetic mirror; MATLAB ; magnetic confinement

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 27 / 150

Design and Installation of Small Angle Slot (SAS) Divertor in DIII-D

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Divertor solutions to efficiently disperse heat from fusion reactors are critical because the maximum steady-state power load is limited to $q_t \leq 5\text{-}10 \text{ MW/m}^2$ to the divertor target. This may pose a special challenge for next-step Advanced Tokamaks (AT), which will have lower plasma density than ITER for high performance long pulse or high duty cycle operations. A new Small Angle Slot (SAS) divertor concept has been developed to address this critical issue. The SOLPS-EIRENE edge code analysis shows that a SAS divertor can achieve strongly dissipative/detached divertor plasmas at a significantly lower upstream plasma density, thus potentially providing a power handling solution for long pulse ATs.

During the vent of the DIII-D vessel in late 2016, a graphite tile SAS divertor was installed. The design of the SAS divertor enables us to test the new slot divertor without affecting the geometry of the existing pumped divertor that is used for high performance advanced tokamak research. The new divertor tiles are mounted to an existing water cooled baffle structure that presently serves as the support structure for the graphite armor tiles for the pumped divertor region. The profile of the new tiles includes a narrow slot that is located outboard of the existing divertor target and divertor pump entrance and does not have any pumping.

Material for the SAS tiles was chosen to be Graftech XTJ-15. XTJ-15 graphite is an isotropic graphite material which is Graftech's replacement for ATJ. ATJ is the material that is primarily used in DIII-D for the graphite armor tiles, but is no longer produced.

This new SAS divertor has been operationally tested during the 2017 DIII-D physics campaign. Special design considerations were required to include Langmuir probes and thermocouples in the slot region of the new divertor.

Details of the tile design, modelling and installation will be presented. Work supported by the U.S. DOE under DE-FC02-04ER54698.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 16 / 151

Thermal Strain Measurement of EAST Tungsten Divertor Module with Bare FBG Sensors

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Tungsten divertor is one of the most important plasma facing components in EAST device. However, it has complex structure and faces extreme work environment. Centralized thermal strain would cause leakages on some weak welding seams, which was harmful for divertor's operation. To measure divertor's thermal strain shall be a valued way to understand its service behavior and then optimize its design and manufacturing process. Fiber Bragg Grating (FBG) was a proven technique to measure temperature and strain in sensor field. Though no similar works have been done before, it was considered an appropriate and feasible method to measure thermal strain on such a high temperature and strong electromagnetic field work environment.

In this work, a heat-resistant bare FBG sensor system had been introduced to measure surface thermal strain of one EAST tungsten divertor module. Ten FBG sensors made in four optical fibers were included in this system. Among the ten FBGs, seven were used for strain measurement and three for strain compensation. A logical compensation method had been adopted to make the results more credible. Heating procedure had been divided into four stages: fast heating stage, slow heating stage, heat preservation stage and cooling stage. Two thermocouples had been used for a feed-back loop

to control heating rate and to record temperature.

The strain measurement system had withstood as high as 210°C temperature and finished the experiment successfully. Experiment results showed that three measurement areas were under tensile strain and four were under compressed strain. Detail strain values had been calculated approximately. The results also had been compared with the measurement results using electric resistance strain gauges. Through the comparison, major results on corresponding areas were found similar, which told that the measurement results were reliable to some extent. In general, areas under tensile strain were more possible to leak than areas under compressed strain. This experiment would be a meaningful reference for tungsten divertor's optimization and maintenance.

Eligible for student paper award?:

Yes

M.OA1: Experimental Devices I / 152

Progress of Interface Design between Test Cell and Lithium Systems in IFMIF-DONES

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The Test Cell (TC) in the IFMIF-DONES (International Fusion Material Irradiation Facility- Demo Oriented Neutron Source) facility is the central confinement to envelop the end section of the accelerator, the lithium Target Assembly (TA), and the test modules. The major functions of the TC include: hosting fusion material irradiation experiments in a leak-tight controlled environment, providing sufficient biological shielding to surrounding rooms against the in-TC neutron and gamma irradiation, and allowing media (mainly lithium and helium) and signal/power penetrations between inside and outside of the TC.

The Lithium System (LS) is connected to its in-TC components through an inlet pipe and an outlet pipe, which penetrate the TC confinement. Two main problems are related to these penetrations: compensation of thermal stresses and minimization of neutron streaming.

The latest TC design optimization has suggested including the lithium collecting tank, the so called Quench Tank (QT), inside the TC, to find a trade-off solution among the simultaneous and conflicting issues of lithium flow stability, irradiation shielding, penetrations into TC confinement, tritium production, and remote handling access.

In this paper, the IFMIF-DONES TC design is updated by introducing a TC-Lithium systems Interface Cell (TLIC) below the TC floor to accommodate thermal stress compensation sections of lithium pipes and irradiation shielding materials. In this configuration, the leak tight boundary of the TC is extended to the inner surface of TLIC through the gaps between the lithium inlet/outlet pipes and main body of the TC floor, and the fixing points of the lithium pipes on the TC boundary is arranged on the wall of the TLIC. Inside the TLIC, lithium pipes will be bended in such a way that the thermal stresses are compensated and direct neutron streaming to the LS area is minimized in combination with removable neutron shielding materials. Preliminary thermal mechanical analysis and neutronic simulations are applied to assist the design of the TLIC and the lithium pipe bends.

The TLIC will be equipped with an air-lock door which can be used as remote handling access to compensation pipe sections and shielding materials during the maintenance periods. Corresponding maintenance scenarios of these components are briefly discussed.

Acknowledgments

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

Eligible for student paper award?:

No

T.OA1: Diagnostics and Instrumentation I / 153

Integration Conceptual Study of Reflectometry Diagnostic for the Main Plasma in DEMO

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The reflectometry diagnostic may present several advantages from the point of view of radiation robustness and components life time as compared to some other traditionally used diagnostics in large fusion devices. From the hardware perspective it does not contain front end elements such as mirrors or sensors which are expected to underperform earlier than the antennas and waveguides when subjected to similar radiation fluxes and deposition/erosion processes. On the other end stresses arising from thermal expansions and electromagnetic (EM) forces can be larger for the waveguides and are accommodated by design. The role of such diagnostic for DEMO is twofold: i) to provide the radial density profile at several poloidal angles (2D map) and ii) to provide data for the feedback control for plasma position. Several groups of antennas need to be distributed along the poloidal section in a number that can satisfy the DEMO control requirements. The study of diagnostic performance and control requirements definition is still being developed and the final number of diagnostic channels is not yet defined, nevertheless several aspects regarding integration can be readily assessed. This paper presents the first case study of integration of antenna groups and waveguides located at several poloidal angular positions covering a full poloidal section of the Helium Cooled Lithium Lead breeding (HCLL) blanket. The integration design shall satisfy strong machine driven constraints (in addition to the physics performance). Diagnostic components installed in the blanket segments must: i) survive for the all period between blanket replacement, ii) be remote handling (RH) compatible with blanket, iii) behave thermomechanical as the blanket structure, iv) cross with integrity the vacuum and reference boundaries (vessel/cryostat/building) and tolerate their relative displacements and v) be compatible with the blanket shielding and cooling services. The present solution developed so far respects several of the main constraints namely, RH compatibility with the full blanket segment and its thermomechanical properties and cooling compatibility but also identifies important issues on the interfaces between the diagnostic antennae extensions and the pipe services at the vessel and also interfaces between vessel and cryostat requiring challenging RH and self-alignment solutions to be demonstrated. Monte Carlo neutronic simulations have been initiated in order to evaluate the heat loads and shielding capabilities of the system. The first results indicate that the cooling for the EUROFER diagnostic components (antennas and waveguides) can in principle be provided by the blanket cooling services (He is considered) via connection to the main Back Supporting Structure (BSS) and routed via the main diagnostic structure body to specific hot spots in the antennas.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 26 / 154**Modeling of pre-Thermal Quench and Thermal Quench stages of disruption induced by Massive Gas Injection in ITER****Author:** Vladimir Leonov¹**Co-authors:** Sergey Konovalov²; Victor Zhogolev³¹ NRC "Kurchatov Institute"² NRC "Kurchatov Institute", Moscow, RF³ NRC "Kurchatov Institute"**Corresponding Authors:** zhogolev_ve@nrcki.ru, leonov_vm@nrcki.ru, konovalov_sv@nrcki.ruModeling of pre-Thermal Quench and Thermal Quench stages
of disruption induced by Massive Gas Injection in ITER

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Abstract

To reduce energy loads on the first wall and divertor during disruption in ITER it is necessary to re-radiate. The ASTRA transport code [2] was used for the modeling of the behavior of the bulk plasma parameters and the Plasma behavior was examined in the stage preceding the thermal quench, pre-TQ, which starts from the appearance of the MGI. Study of the operational space of the ITER DMS system based on a MGI with use of different noble gases and

- The dependence of the pre-TQ and TQ stages durations on the impurity species and quantities are calculated.
- Assimilation coefficients for the different gases and mixtures are presented. Physical mechanisms hampering impurity penetration into the pre-TQ plasma at MGI are discussed.
- The dependence of the radiated power on injected gas quantity during stages under consideration is presented.
 - The amount of different gases necessary to re-radiate of more than 90% of the energy content of the plasma column in the pre-TQ and TQ stages are estimated.

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Eligible for student paper award?:

No

R.OP6: Safety, Operations, and Maintenance / 155

CorteX: A Standardised Remote Operations Communications System that is Inherently Designed to Accommodate Change

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Robotic remote operations are commonly used to conduct necessary work such as maintenance, decommissioning and experimental reconfiguration in environments that are hazardous to humans. This is especially prevalent within the nuclear fusion world, where devices such as JET, ITER, DEMO, and CFETR do or will require the remote execution of robotic remote operations of significant number and complexity. This poses many challenges, including the need to integrate many different bespoke and COTS systems from a variety of international suppliers, the need to address ongoing changing operational requirements, and through-life issues such as obsolescence management and protection. It is clear that it is advantageous to modularise, encapsulate, and decouple these systems to maximise efficiency, maintainability, and value.

We present a new communications protocol and software framework that begins to address many of these issues. CorteX is a distributed, homogeneous architecture for integration of devices, software, and intelligence within a complex remote operations environment, the core of which is based around a communications standard that is presented. In conjunction with a suitable hardware design, this framework provides the means to allow systems to quickly and seamlessly integrate and interoperate. It is explicitly designed to accommodate foreseen and unforeseen changes to the overall remote maintenance capability whilst minimising the impact of those changes on other subsystems and on the operations team. Inspiration has been taken from the experiences of operating and modifying a remote handling system at JET over the course of several decades.

The communications standard will be presented along with a new compatible software framework that has been implemented. Initial systems integration and operational trials conducted in a laboratory environment in order to determine and demonstrate capability will be presented for the first time. This includes integration of a complete end-to-end remote handling system incorporating tens of devices including a variety of robotic systems similar to those seen in fusion and other remote maintenance environments, as well as various software systems such as Command and Control interfaces and Virtual Reality synthetic viewing systems.

We will describe future direction of the work including further integration of the system into various environments including work related to JET, ITER, DEMO, and the European Spallation Source (ESS).

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 17 / 156

EM Analysis of ITER Diagnostics Upper Port Plugs 14 (US port) and its in-Port components during Plasma Disruptions

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ITER diagnostic port plugs perform many functions including nuclear shielding, structural support of diagnostic system, while allowing for diagnostic access to the plasma. With design advancing, the in-port diagnostic components are integrated into the port plug structure, and the diagnostic shield modules (DSM) are customized to accommodate various in-port diagnostic components. This paper summarizes results of transient electro-magnetic analysis using Opera 3d in support of recent design activities for ITER diagnostic upper port plug 14 (UPP14). A complete distribution of disruption loads on each component in UPP14 is presented. Impacts of different design features, such as the locations of the electrical contact, to the EM loads are discussed, and the solutions for improving the port structure are proposed.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 18 / 157

Development of primary vacuum windows for ITER diagnostics

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Most of ITER's diagnostics will be provided with viewing lines (optical, microwave, spectroscopic) for the monitoring of key characteristics of the plasma or for the achievement of physical measurements inside the vacuum vessel. The nature of the physical signal transmitted through the viewing lines requires the implementation of window assemblies incorporating non-metallic window. Placed at the vacuum boundary, the window assembly shall also ensure the vacuum integrity required for the plasma. Moreover, the diagnostic window assemblies form part of the ITER primary confinement boundary. Their integrity is consequently of prime importance in containing the reactant materials such as tritium in the inside of the vacuum vessel and, thus, directly related to the Nuclear Safety. The window assemblies are Protection Important Components (PICs) and their design, procurement and operation are considered as Protection Important Activities (PIAs). The primary confinement boundary shall be fully ensured during all the normal and accidental conditions. Window assemblies are part of the unpressurized area of the radioactivity confinement barrier, with safety related functions. These assemblies are made from components with non-metallic materials, particularly high grade ceramics, which are generally not covered by pressure vessel codes and for which there is no existing industrial standard that specifies the criteria for the design, manufacturing, and testing directly applicable to ITER. To incorporate non-metallic replaceable window assemblies in the confinement barrier of ITER, it shall be a requirement that the Operator (ITER Organization) has these window assemblies designed, procured, installed and operated based on procedures and records the French Nuclear Order known as "the Order 7th February 2012", concerning basic nuclear installation design construction and operation quality.

The paper will discuss the progress in the ongoing design and development of the primary vacuum window assemblies, covering several aspects, such as design, interfacing loads, integration in the port infrastructure and impact on the diagnostic performance. Also, maintenance and associated tools for window assemblies will be discussed.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 19 / 158

Real-time Two-dimensional Optical Polarization Properties of the Fusion Reactor First Mirror Based on Active Polarized Beams

Authors: Junli Qi¹; Jinhong Yang²; Haifei Deng²; Weihua Wang³¹ *University of Science and Technology of China*² *New Star Institute of Applied Technology*³ *Institute of Applied Physics of AOA***Corresponding Authors:** yjh_pbxy@sina.com, whwang@ipp.ac.cn, kinger1104@163.com, qijunli_r@163.com

With the construction of the next generation of large size Tokamak marked by ITER and the realization of steady state plasma discharge, the first mirror(FM) will become the core of optical elements which will be used in the optical diagnosis system of future fusion reactor. Under the action of plasma, the FM surface morphology, structure and composition, and optical properties will be changed, which seriously affect the optical performance and service life of the FM. FM problem, directly related to the effectiveness of the optical diagnosis system, is an extremely important research topic of the future plasma physics diagnosis. At present, the FM research on optical properties mostly stays on the reflectivity, the polarization properties in nature are little studied. Besides, most researches on FM are performed on a point and off-line detection, and the use of polarized beams to carry out real-time on-line two-dimensional detection of the FM are less. In this manuscript, based on the polarization characteristics of a metal object, the experiment system of active polarized beam detecting FM optical polarization properties was established. The two-dimensional polarization information varieties of the same incident beam reflecting from different FMs, including the degree of polarization (DOP), polarization azimuth (PA) and circular polarization angle (CPA), were in real-time studied. It is found that the two-dimensional DOP and CPA can be used to detect different roughness degree Ra and regular degree of the measured surface, and provide significant basis for further quantitative detection of FM damage degree.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 7 / 159

Study on considerable defects introduced in tritium breeding material Li₂TiO₃ by annealing in vacuum

Authors: Qiang Qi¹; Jing Wang¹; Maoqiao Xiang²; Yingchun Zhang²; Guangnan Luo¹¹ *Institute of Plasma Physics, Chinese Academy of Sciences*² *School of Materials Science and Engineering, University of Science and Technology Beijing***Corresponding Author:** qiqiang@ipp.ac.cn

Li₂TiO₃ is one of the most promising candidates for solid breeder materials. However, defects introduced in Li₂TiO₃ will affect tritium release. In the present study, vacuum-annealing defects in Li₂TiO₃ were investigated by means of electron spin resonance (ESR). The defects of E-centers were found to be introduced by vacuum-annealing in Li₂TiO₃. The color of Li₂TiO₃ samples becomes dark grey after annealing in vacuum. This color change suggests the change from Ti⁴⁺ to Ti³⁺ due to

decrease in the oxygen content. And the color was observed to recover to initial color, white again after annealing in air. The concentration of vacuum-annealing defects reaches almost a constant when the pressure is lower than 10 Pa. The defect concentration increases as annealing temperature goes up and then decreases when the temperature reaches to a certain value. The amount of vacuum-annealing defects goes down and the color of Li_2TiO_3 samples recovers to white gradually when vacuum-annealing samples annealed in air at different temperatures. There are no defects and color change while Li_2TiO_3 samples anneal in air first and then transfer to vacuum tube rapidly for vacuum-annealing. More defects were introduced in Li_2TiO_3 samples immersed in water for 6 hours. This elucidates that the defects produced by vacuum-annealing are attributed to the reduction of water adsorbed in Li_2TiO_3 . Mass of Li_2TiO_3 was found to vary after the change of the atmosphere from nitrogen to air investigated by thermogravimetry. X-ray diffraction (XRD) results indicate that there are no modifications on Li_2TiO_3 crystal phases.

The authors acknowledge School of Materials Science and Engineering in University of Science and Technology Beijing for providing the experimental samples. This work supported by the National Natural Science Foundation of China under contract No. 11605230.

Eligible for student paper award?:

No

M.OP2: Materials I - Board: 8 / 160

Design, synthesis and characterization of Li_4SiO_4 -based solid solutions as advanced tritium breeders

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The breeding blanket is a key component of the fusion reactor since it involves tritium breeding and energy extraction, both of which are critically important for the development of fusion power. Different lithium based ceramics have been studied as attractive tritium breeder materials, Li_4SiO_4 has been selected as one of the most promising candidates for solid tritium breeding materials in fusion reactors because of its high lithium atom density, its high melting temperature and favorable tritium release behavior. Li_4SiO_4 -based solid solutions: $\text{Li}_{4+x}(\text{Si}_{1-x}\text{Al}_x)\text{O}_4$ and $\text{Li}_4\text{Si}_{1-x}\text{Ti}_x\text{O}_4$ were prepared as advanced tritium breeder to improve the mechanical property, irradiation resistance and reduce the tritium retention. Different Li_4SiO_4 -based solid solutions powders and pebbles containing aluminum and titanium were prepared by solid state reactions and Modified melt-spraying process. Phase analysis, microstructures and density of the ceramics were determined by XRD, SEM and Archimedes' method. Impedance spectroscopy was measured to evaluate the electrical conduction properties of the ceramics. The thermal conductivity was determined using a laser flash device. Tritium release performance in $\text{Li}_{4+x}(\text{Si}_{1-x}\text{Al}_x)\text{O}_4$ and $\text{Li}_4\text{Si}_{1-x}\text{Ti}_x\text{O}_4$ irradiated with thermal neutron was studied by out-of-pile annealing experiments. These facts would represent the following advantages to use Li_4SiO_4 -based solid solutions in blanket system of D-T fusion reactor that the thermal conductivity is higher and tritium inventory is lower in Li_4SiO_4 -based solid solutions than those in Li_4SiO_4 .

Keywords: $\text{Li}_{4+x}(\text{Si}_{1-x}\text{Al}_x)\text{O}_4$, $\text{Li}_4(\text{Si}_{1-x}\text{Ti}_x)\text{O}_4$, thermal conductivity, the mechanical property, tritium release performance

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 9 / 161

The numerical simulation for the heat transfer enhancement experiments of the HCCB-TBM first wall

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The first wall (FW) of helium gas Cooled Ceramic Breeder (HCCB) Test Blanket Module (TBM) for ITER need bear the loads like high power density heat flux from plasma, nuclear heat from neutron deposition on the structure, and the transient high heat loads like plasma disruption. The average heat flux density on FW of HCCB-TBM is about 0.3MW/m² and the maximum partial transient heat flux may reach up to 1MW/m². In current design schemes in ITER, Reduced-Activation ferritic/Martensitic (RAFM) steel is mostly selected as structural material, several groups of "radial-circular-radial" flow channels are placed inside FW, and 8MPa high pressure helium gas is applied as coolant to remove the heat flux on the surface and the nuclear heat by neutron deposition. As for smooth channel, required heat transfer efficiency and structural security can only be achieved after the flow velocity of helium gas reaches 50~80m/s (operating temperature of RAFM steel structure is lower than 550°C). High flow velocity of helium gas consumes a large amount of pumping power which lowers the net output power of reactor and increases greatly the equipment cost. Although roughness (less than 10µm) technique on the flow channel surface enhance heat transfer efficiency to some extent, the average heat transfer coefficient increases by less than 10% (from 2700 W/m²K to 2900 W/m²K). To enhance the helium gas cooling efficiency and security in FW, heat transfer enhancement technology needs improving and optimizing for the design scheme of helium flow channel to meet the functional requirements of FW. Based on this objective, the filling-evacuating HPHCL (High-Pressure Helium-Cooled Loop) were build to test and prove the heat transfer enhancement schemes of helium gas cooling FW. In this paper, the design scheme of the filling-evacuating HPHCL is presented, and the key issues of engineering manufacture and the test cases are calculated and analyzed. As for the first step, the CFD numerical simulation method is adopted to simulate the test cases of filling-evacuating HPHCL. The sustainable evacuating time under different mass flow rate of the He gas are estimated. On the basis of calculating helium gas cooling scheme of FW smooth flow channel, FW structural temperature gradient, maximum wall temperature, average heat transfer coefficient, and pressure drop of flow channel are selected as evaluation indexes. Three dimension numerical simulation results are compared to acquire optimizational heat transfer enhancement schemes like placing transversal ribs and V-shaped ribs in the flow channel of front wall of FW. The helium gas turbulence intensity and the heat transfer area are improved through optimizing the distance and angle between V-shaped ribs and other coefficients to enhance heat transfer. The calculation results are used as reference for the next verification experiments, which the 8~10MPa high pressure helium gas will be selected as coolant.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 45 / 162

Optimized Shape of TF Coil

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The effect of gravity is Insignificant on the Conductor tension of TF coil. For The Chinese Fusion Engineering Test Reactor (CFETR), the influence is about one ten thousandth. Therefore, the design of TF coil shape generally does not consider its gravity.

The shape of the TF coil is determined by the constant-tension and without bending moment equation regardless of its gravity. By calculating the equation, the Princeton-D curve can be worked out in the case of giving the radial position of the TF centerline. Because the radius of curvature of Princeton-D curve varies continuously in space, it is difficult to manufacture. Therefore, the Princeton-D curve is generally fitted with three symmetrical arcs up and down.

There are four variables in the fitting process, and the fitting result may be not optimal because of different fitting methods. In this paper, a criterion is given to determine whether the fitting result is optimal or not. Based on the criterion, it is possible to determine whether the shape of TF coil is optimal or not for each fusion device.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 20 / 163

Improvement of the plasma current density profile by the polarimeter/interferometer system on the EAST tokamak

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A novel method has been developed to improve the accuracy of plasma current density profile by combining the polarimeter/interferometer (POINT) measurements with the external magnetic measurements on the EAST tokamak. The POINT system, measuring the accurate line-integrated electron density and Faraday rotation angle, provides the magnetic field information inside the plasma. By adding these data to the equilibrium confinement, the results from POINT measurements show a difference with the original equilibrium and the difference becomes larger from boundary to core of the plasma. This correction process makes up for the deficiency of magnetic probe measurements, the details of the correction process are specified, which bypass the equilibrium fit (EFIT) code. Results with and without these corrections are presented, comparisons of the corrected results and experimental results are also shown and they are found agree well with each other. The feasibility and reliability of the correction process are also discussed in this paper.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 72 / 165

The in-vessel protection components for ITER First Plasma operation

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ITER plasma operations begin with a limited ‘first plasma’, with the main purpose of concluding integrated commissioning of the tokamak. This first plasma is targeted to last at least 100 ms and with a minimum plasma current of 100 kA. As in other tokamaks, the first plasma is conducted prior to full installation of the baseline in-vessel components. Namely, the blanket and divertor are installed only after first plasma experiments are completed. To avoid potential adverse effects of start-up plasma on the vacuum vessel and other already-installed high-value components (in-vessel coils, cable looms, etc.), several temporary in-vessel components are installed to protect all in-vessel systems that cannot tolerate direct plasma interaction.

Machine protection is in part provided by temporary limiters and divertor replacement structures, which together create a poloidal and toroidal guide to shelter the vacuum vessel from the plasma and possible fast particle beams. 72 temporary limiter structures are distributed around the tokamak as four poloidal loops of eighteen tiles that follow the first wall contour. For engineering safety during plasma commissioning, the system is capable of maintaining plasma pulses of up to 3 seconds, tolerating up to 30 MJ of thermal energy and 1 MA of plasma current. Divertor replacement structures complete the bottom periphery of the poloidal loop to interrupt any potential downward plasma movement.

In addition to protection from the plasma, the vessel also requires protection from the electron cyclotron resonance heating (ECRH) beam. This beam injects 6 MW of microwave energy across the null region of the vacuum vessel to help energize and assist the breakdown of first plasma. To protect the vacuum vessel, an inboard mounted mirror reflects the ECRH beam energy from its upper port origin into an outboard equatorial port, where a second component (a beam dump) can effectively absorb the radiation.

These first plasma protection components (FPPC) are unique to ITER, and fit the need of a simple and cost-effective system that is easily installed and un-installed prior to further plasma operation in such a way as to minimize later impact on machine operation. The initial FPPC effort matured these components to a conceptual design level in preparation for the Conceptual Design Review held in November 2016. This paper summarizes the FPPC design and analysis status resulting from this initial effort.

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

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 78 / 166

Mechanism for Plasma Fusion with Major Ionic Species at Only Ten Million Kelvins

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Ten-thousand-fold enhancements of ³He abundance in impulsive solar energetic particle (SEP) events indicate that ³He ions, before accelerated to high energies (~ 1 MeV/nucleon) in solar flares, should

have been preferentially heated significantly ($T_{3He}/T_{4He} 10 - 100$). The best mechanism developed so far for preferential heating of 3He is the first or second harmonic resonances of 3He with current-driven electrostatic 4He or H-cyclotron waves. This mechanism for minor ionic species to be strongly heated by major ionic species cyclotron waves may also play an important role in the heating of laboratory plasmas for nuclear fusion. It is well known that nuclear fusion is a high-energy reaction in which two extremely energized lighter atomic nuclei, after overcoming the sturdy Coulomb barrier, fuse tightly into a heavier one via the nuclear force. For the fusion to occur among nuclei in labs such as in a tokamak, the plasma must be heated to or above 100 million Kelvins (MK) with sufficient confinement time and sufficient plasma density. Efforts on experiments of plasma fusion conducted in the past decades indicate numerous mysteries and difficulties surrounding how to control heat bursts and confine plasmas of extreme temperatures. Based on our previous work for 3He -rich SEP events, we propose a new general mechanism for plasma fusion of minor ionic species extremely heated with major ionic species at only about 10 MK in order to reduce difficulties of fusion technology and engineering in the plasma confinement and control. We consider multi-ion plasmas composing of various major and minor ionic species with a current drive. As an electric current is driven through, a plasma can be ohmically heated by the current up to 10 MK, at which the resistivity in the plasma is too low for the current to be significantly dissipated further and the entire plasma saturates its temperature at this level in this first-stage of the heating process. When the current is continuously driven up to a critical point, e.g. thirty percent of the electron thermal current, electrostatic ion-cyclotron waves of the major ionic species are destabilized, which can have frequencies at around a multiple of the ion-cyclotron frequencies of the minor ionic species and thus can further heat the minor ionic species via particular harmonic resonances to 100 MK and higher, at which the nuclear fusion between the extremely heated minor ionic species and the relatively cold major ionic species can occur. In this second-stage of the heating process, only the minor ionic species are preferentially heated. This mechanism of plasma fusion, because temperatures of the major ionic species and electrons are only around 10 MK, can greatly reduce difficulties of technology and engineering in confinement and control of the fusing plasma.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 46 / 167

Insulation Systems for the ITER Central Solenoid Modules

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General Atomics is currently fabricating seven (six plus one spare) modules for the ITER Central Solenoid in its Poway, CA facility. One of the critical steps in developing the fabrication process for the modules was to develop the insulation system providing the electrical isolation between turns and to grounded components. The insulation system for a module was required to withstand a maximum test voltage of 30kV, with a design goal of 150kV to ground for all penetrations and leads. The complex geometries of the coil, the helium penetrations, leads, and helium piping required the use of novel materials and approaches for the insulation.

Materials were developed to improve the handling of large sheets of fiberglass and Kapton® polyimide of the ground insulation. On the vertical surfaces and around the corners of the CSM, sheets of lightly-bonded Glass-Kapton®-Glass (GKG) and Glass-Kapton® (GK) were used. These sheets were developed and tested for electrical strength and resin permeability in comparison to glass and Kapton® sheets.

Around the leads and helium penetrations, electrical strength and tracking distance were obtained using built-up layers of a thermoplastic polyimide (TPI). Sheets of DuPont Mitsui AURUM® TPI were thermoformed and interlaced with other sheet materials to obtain the required tracking distance.

The liquid helium supply and return pipes were insulated with 20 layers of Kapton® coated on one

side with a B-stage epoxy resin that was dry to the touch. These Kapton® tapes were wrapped around the pipes (without interstitial layers of glass), with a ground mesh and a final outer layer of prepreg glass for durability, and then cured. The insulation was tested for electrical strength in air and in Paschen conditions.

Instrumentation wires attached to the conductor exited the ground insulation along the helium pipes. The sealing of the “cusps” between the round wires and the round pipes was accomplished with high resin content prepreg glass. A methodology was developed to eliminate a problem of wire insulation cracking after curing the epoxy. These wire exits were tested in Paschen conditions.

A ground screen was installed around the entire coil, its penetrations, and the helium pipes. The ground screen consisted of stainless steel mesh sheets spot-welded together with a single-point ground. The ground screen for the leads and helium penetrations was made by forming the screen pieces and connected to the pipe ground mesh with spot welds.

A qualification coil was insulated and tested to 10kV prior to resin injection. A vacuum pressure impregnation of the resin was completed after which the coil passed a 30kV hipot. This qualification coil serves as a final test article to validate the design and fabrication of the insulation system.

This paper describes the insulation system and the development and testing of the novel insulation materials used in the ITER CSM.

Acknowledgement: This work was supported by UT-Battelle/Oak Ridge National Laboratory under sponsorship of the US Department of Energy Office of Science under Awards 4000103039 and DE-AC05-00OR22725.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 47 / 168

Test Results of ITER 52-kA HTS Current Lead Prototypes

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The Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP) is responsible for the construction and performance testing of the 30 pairs of ITER high-temperature superconducting (HTS) current leads. A first pair of Poloidal Field (PF) coil type, 52-kA HTS current lead prototypes was built and cold tested in ASIPP in mid-2016. The test results approved their excellent performance on low joint resistance, long loss-of-flow accident time and high current-sharing temperature. The overheating time, mass flow, and heat loads to 5-K ends also meet the expectation. This paper summarizes the major test results for the PF 52-kA HTS current lead prototypes.

Eligible for student paper award?:

No

T.OP3: Project Management and Systems Engineering / 169

Preparation of ITER Tokamak Assembly and Tooling

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ITER starts assembly! What sounds simple is actually an enormous technical challenge for the ITER Organization in the coming years. Since 2016, after the design and manufacture phases of Tokamak components, the project is now entering the assembly phase step-by-step.

A new organizational structure was put into place to start the preparation, installation and commissioning of all components and systems. This shall be done in close collaboration with all Domestic Agencies, Construction Management as Agent (CMA), IO Works Contractors and IO staff. The established Construction Team is subdivided into three groups, taking responsibilities in different installation worksites.

1. Construction Team for Tokamak Assembly (CTTA)
2. Construction Team for Tokamak Complex (CTTC)
3. Construction Team for Plant Installation (CTPI)

Furthermore, the CMA is actively supporting the three Construction Teams for the preparation, coordination and supervision of assembly activities. The final execution of the work shall be performed by Works Contractors, i.e. industrial companies / consortia with adequate experiences in the field for their assignments. This set-up facilitates the need for a smaller work-force initially, and a fast ramp up of resources in later assembly phases.

Primary focus is currently given to the definition and planning of the Construction Work Packages to be executed within the next 2 years; identification and procurement of assembly tools, qualification of processes and tendering of the major installation contracts.

The Tokamak Assembly preparation will have its on-site commencement in 2017 with the arrival of the first Sector Sub-Assembly Tool. This is a major tool which will assure the Sector Sub-Assembly of three major components: the vacuum vessel sector, the associated vacuum vessel thermal shield and a pair of toroidal field coils. This In-Kind tool is supplied by KODA and will be first assembled and tested at the factory before shipping to IO where it will be re-erected in the IO assembly hall from third quarter this year. After site acceptance, the test campaign will continue until final commissioning to meet the regulations applicable in France. Manufacture of other major tools such as sector lifting, upending and in-pit assembly tools as well as Cryostat handling tools will also commence this year.

One example of the above mentioned assembly process qualification is the comprehensive test of welding processes on 1:1 mock-ups of vacuum vessel sectors and ports. This is of special interest as the vacuum vessel is a "Protection Important Component". These qualification activities will help to ensure compliance with nuclear safety requirements.

The presentation will summarise the main Tokamak Assembly preparation features and will touch on related organizational, tender and safety aspects.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 81 / 170

Fast Boundary Reconstruction from Tangentially Viewed Visible Images for Plasma Control in EAST

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The fast plasma boundary reconstruction is usually used for real-time control of tokamak plasma. In EAST experiment, the time consuming for boundary reconstruction should be within 1ms to meet the need of real-time control. A fast algorithm based on visible imaging diagnostics is developed in EAST to reconstruct the plasma boundary directly and independently. Compared to the results of EFIT, the overall average error is within 1.5cm, the average error at the lower X point is within 1cm, and the average error at the outermost and innermost points of LCFS are below 0.5cm. The causes of the deviation are discussed, and the methods for decrease are presented. For an image with the size of 680×544, the algorithm implemented by C++ with OpenCV can complete the computation in 0.9ms, achieving an acceleration of 300 times, when compared with parallel MATLAB. Furthermore, when the pixels of camera sensor is not saturated, the algorithm is robust for different intensities of the discharge images.

Keywords: Plasma boundary reconstruction; Visible imaging diagnostics; EAST; OpenCV; Plasma control

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 82 / 171

Analysis of non-inductively high-performance discharges

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EAST research program aims at achieving steady-state long-pulse operations, which have been obtained with fully non-inductively current drive and heating, maintaining around zero loop voltage for nearly the entire plasma current flat-top in about 60s at EAST shot #67341 recently. Based on the analysis of non-inductive current fractions, high β_p is desirable in order to sustain steady-state high performance discharges on EAST in the near future. The effect of bootstrap current relates to the nonlinear component of vertical magnetic field judged by Maxwell equations. Furthermore, the quasi-linearity relationship in flat-top phase between vertical magnetic field and line-averaged plasma density lays the theoretical basis for radial compression. An increase in magnetic strength will allow high density, high beta, high bootstrap current fraction and high fusion gain to be reached, offering an attractive regime for compressed plasma to approach the Lawson parameter, especially for steady state operation of the designed CFETR—Chinese Fusion Engineering Testing Reactor. Existing limitations of EAST tokamak are analyzed for accommodating and simulating the high-performance discharges.

Index Terms—EAST tokamak, non-inductive current drive, high beta, vertical magnetic field, compressed plasma

Eligible for student paper award?:

Yes

R.OP1: Diagnostics and Instrumentation II / 172

Estimation of X-mode reflectometry first fringe frequency using neural networks

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One of the main issues with X-mode density profile reflectometry is the correct determination of the upper cut off first fringe reflection.

The propagation of electromagnetic waves in X mode in plasmas is characterized by two cut off frequency regions that depend on the local plasma electron density and the local magnetic field.

By using the upper cut off region for electron density profile reflectometry group delay measurements, the plasma can be probed from near zero density, $n_e \approx 0$, up to a maximum density limited by the probing frequency band.

However, due to the broad operational conditions of an experimental nuclear fusion reactor, the first fringe plasma reflection can occur at any probing frequency.

The first fringe reflection is used together with the magnetic field profile to determine vacuum distance between the reflectometer antenna and the start of the plasma. An incorrect estimation not only introduces a radial error but also a group delay error, affecting the shape of the resulting density profile.

A new multichannel X-mode density profile reflectometry diagnostic was recently installed on ASDEX Upgrade 1 to study the edge plasma layers in front of the ICRF antenna by probing the plasma in the 40-68 GHz range. In this work we study how the different operational conditions of the fusion reactor affect the reflectometry raw measurements. We take advantage of the high signal-to-noise ratio of the diagnostic to determine a few raw signal features that are consistent with the observation from the SOL, where the plasma starts to be optically thick. We apply these features to develop an algorithm to estimate and track the variation of the first fringe frequency along a discharge.

Tests show that the algorithm is able to accurately determine the first fringe for most discharges. However, for a small number of unanticipated cases, the algorithm fails, introducing inevitable errors in the density profiles.

We present a new algorithm that uses a neural network approach for the first time for estimating the frequency of the first fringe. A varied training set using similar decision features was carefully selected by experienced reflectometry diagnosticians and used to train the neural network model using the open source software library TensorFlow [2].

Early results show that the neural network model is able to predict the frequency of the first fringe to within the degree of uncertainty of the human diagnostician. The comparison of results using both algorithms is presented. Both algorithms shall be used together to minimize the error of first fringe estimation.

Eligible for student paper award?:

Yes

M.OA2: Divertors and High Heat Flux Components / 173

Virtual Engineering of a fusion reactor: application to divertor design, manufacture and testing

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Owing to ever-growing computational power, Virtual Engineering is transitioning from a possibility to an absolute necessity, and has exciting potential to accelerate the realisation of a fusion power reactor. Virtual Engineering is the use of sophisticated computational modelling to enhance the traditional route of component qualification by providing deeper insight, shortening the design cycle, reducing the burden of costly experimental testing or by answering questions that are simply not possible to answer by testing or real-world measurement. Finite element analysis (FEA), has been used for decades to perform engineering calculations and support design substantiation. In Virtual Engineering, the execution of FEA is highly parallelised, and mathematical optimisation is used to efficiently explore the design space, dramatically reducing design time. Further, typical FEA uses a much simplified and idealised representation of a real component, often taking input from computer aided design models. The advent of high performance computing and 3-D volumetric scanning offers the capability to create a highly accurate virtual “twin” of an as-manufactured component, including any unintended features or imperfections. These virtual components can be virtually tested under realistic conditions to simulate manufacturing, assembly, commissioning or operating phases.

In this work, the potential of Virtual Engineering in the design and validation cycle is demonstrated for the first time in an application to DEMO water cooled divertor target design, manufacture and testing. First, FEA-based design search and optimisation is used to improve the established tungsten monoblock divertor concept. Fabricated mock-ups are imaged using X-ray tomography and analysed using image-based finite element modelling (IBFEM) to simulate the in-service high heat flux conditions. The IBFEM captures imperfections in the manufacturing process; in the example here an incomplete braze between components of the monoblock mock-up. The response of the FEA model to in-service simulations of heat flux indicated that the stress within the monoblock exceeded design limits leading to mechanical failure. A revised manufacturing procedure was introduced to improve the joining procedure and eliminate voids. The in-service simulations can be used to investigate the impact of an imperfection not only to identify early failure but also to identify acceptable imperfections that do not compromise lifetime performance, avoiding unnecessary component rejection and thus reducing cost and time to realisation. Mock-ups fabricated to the improved joining procedure have been successfully tested by infrared thermography, and under high heat flux at up to 25 MW/m² with no sign of damage.

1 T. Hirai et al., Fusion Engineering and Design, 88 (2013), p1798-1801.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 1 / 174

RIPER: An irradiation facility to test Radiation Induced Permeation and Release of deuterium for fusion reactor materials.

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For successful future Fusion Power Plant operation, tritium self-sufficiency is an essential element of the multiple technical challenges facing the fusion programme. In particular all the different types of candidate blankets will have to make use of different functional materials, such as SiC for flow channel inserts (FCI), ceramic coatings on steel for liquid metal blankets, and Li-ceramic breeders for the helium cooled pebble blanket (HCBP). For all these advanced materials radiation enhanced tritium

permeation and retention are of concern. Also thermochemical, and in particular radiation stability, must be taken into account. Radiation induced changes in composition and microstructure may alter either production and/or extraction of tritium, permeation, and retention, hence seriously affecting the achievable tritium breeding ratio (TBR). The validation of these advanced materials requires experimental data to be obtained under as near as possible reactor operating conditions.

At CIEMAT (Research Centre for Energy, Environment, and Technology) during the various Euratom and Broader Approach (BA) agreements, different experimental systems have been developed in the beam line of a 2 MV Van de Graaff electron accelerator. These allow one to study in situ numerous radiation enhanced and induced effects such as electrical, luminescence, and diffusion properties in fusion insulating and breeding blanket materials. Within this framework the Radiation Induced Permeation and Release (RIPER) facility has been developed to provide essential tritium related data. The facility consists of several special irradiation chambers and corresponding experimental systems to measure deuterium permeation for ceramic coated metals during irradiation at different irradiation temperatures and gas pressures, as well as deuterium release from Li-ceramic breeders during irradiation. In the same beam line facility it is also possible to measure hydrogen isotope (H and D) transport under relevant conditions, where deuterium adsorption, absorption, and desorption, including thermally induced desorption (TID) can be measured under controlled ionizing radiation and temperature conditions. The system also allows one to determine possible decomposition such as lithium vaporization/release during irradiation and/or heating of the Li-ceramic materials. All the above gas release processes are monitored using a Pfeiffer Smart Test commercial gas leak detector (He, D₂ sensitivity ≥ 10 -12 mbar.l/s) and a Pfeiffer PrismaPlus QMG 220 residual gas analyser –quadrupole mass spectrometer (sensitivity ≥ 10 -14 mbar) connected to the vacuum system.

The paper will give a detailed description of the above experimental systems as used to test ionizing radiation and temperature effects on the functional properties of candidate breeding ceramics, silicon carbide for FCIs, and radiation induced permeation through alumina coated stainless steel.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 66 / 175

Assembly methodology and tools developed for Tore Supra transformation into WEST platform

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Since 2013 Tore Supra has turned to WEST (Tungsten (W) Environment in Steady state Tokamak) Platform targeted at supporting the ITER divertor detailed design, manufacturing and future operation. The major changes included the modification of the magnetic configuration (from limiter to divertor), the replacement of carbon plasma facing components by new tungsten plasma facing components (PFCs), the upgrade of the high frequency heating systems and diagnostics. This resulted in replacing 100% of the inner components and about 80% of port-plugs components.

The main technical challenges consisted in:

- Assembling interlinked new elements in the existing device;
- Designing the in-vessel interfaces and accurately positioning the components in order to maximize the plasma volume;
- Designing and in-situ manufacturing of the divertor coils.

The key phase of the assembly sequence was the in-situ Divertor coil construction, which required developing specific techniques such as in-situ brazing, wrapping, etc...and controlling perfectly

operations as no repair is possible after the divertor structure closing. The assembly work has been organized in several steps before and after the in-situ Divertor coil construction.

Sub-millimeter metrology was key to provide input data on the existing environment, to transfer the magnetic references from Tore Supra to WEST and also accurately position the elements. In addition, the CEA virtual reality room was widely used for kinematics definition and, in a second step, for generic tooling validation.

The paper will describe the main sequences defined for WEST assembly and associated qualification processes applied before and after component installation. Lessons learned in terms of design to assembly and associated tooling cycle will be detailed.

Eligible for student paper award?:

No

T.OP3: Project Management and Systems Engineering / 176

Extent of Condition Review of the NSTX-U Project

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The National Spherical Torus eXperiment Upgrade (NSTX-U) is an experimental research facility funded by the U.S. Department of Energy (DOE) Fusion Energy Sciences (FES) that is operating at the Princeton Plasma Physics Laboratory (PPPL). NSTX-U (<http://nstx-u.pppl.gov/home>) is the centerpiece of the U.S. ST research program. After commencing operation in 1999 in its original configuration, the NSTX device operated successfully for a period of 10 years and served as a proof-of-principle demonstration of the low-aspect-ratio ST concept.

An upgrade initiative commenced in early 2009, aimed at improving the understanding of the ST configuration and establishing the physics basis for next-step ST facilities. In particular, operation at higher magnetic field with reduced plasma collisionality is targeted by the upgrade. Controllable fully non-inductive current-drive will also contribute to assessment of the ST as a potentially cost-effective path to fusion energy.

Per the United States Department of Energy (DOE) order DOE O 413.3B the Critical Decision 0 (CD-0) Mission Need for the NSTX Upgrade Project was approved February 2009. The CD-4 Project Completion milestone, achievement of 1st plasma, was accomplished in August 2015.

During the early phases of commissioning and operation, a series of technical problems were encountered, the last of which involved the failure of one of the poloidal field coils due to a turn-to-turn fault. This failure necessitated a shutdown of the NSTX-U device in July 2016. Soon after this event the DOE directed PPPL to conduct an Extent of Condition (EoC) review to “identify all design, construction, and operational issues”. And to “Prepare a corrective action plan (CAP) to include cost, schedule, scope, and technical specifications of action.”

In response to this directive, PPPL organized a dedicated Recovery Project team with Responsible Engineers linked to 11 subdivisions of the project scope. For each of those subdivisions, plus another covering the overall project scope and requirements, special Design Verification and Validation Reviews (DVVRs) are being convened to flesh out any gaps or issues in the design basis or as-built configuration of the device and its supporting infrastructure. The results from these DVVRs will feed into the CAP. Each line item in the CAP will address the issue, its consequence, the cost/schedule to mitigate, and the post-mitigation condition. The information in the CAP will then feed into high-level programmatic decisions concerning the path forward.

This paper describes the background leading to the EoC directive, the DVVR process, and the CAP process.

*This work is supported by US DOE Contract No. DE-AC02-09CH11466

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 50 / 177

The enhancement of high temperature deformation resistance for V-4Cr-4Ti alloys

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Vanadium alloys, especially those with the compositions of V-4Cr-4Ti, are important candidate materials for blanket in future fusion reactors. In the past one decade, to enhance the high temperature mechanical properties, which equates with safely increasing the operation temperature, is one of the main efforts made to V-4Cr-4Ti alloys. According to dislocation theory, to properly increase defects density is an efficient way to strengthen practical alloys. Moreover, for high temperature application, the thermal stability of such defects is required.

This work presents various commercial techniques used for strengthening the V-4Cr-4Ti alloy, including alloying, cold work, aging and mechanical alloying (MA) as the highlighted topic. With characterization of the tensile and creep properties at elevated temperatures coupled with investigation on deformation mechanisms of the resulted V-4Cr-4Ti materials, more work is then focused on the nano-particle dispersion strengthening.

In the experiments, different starting powders, carbide dispersion agents and MA routes are used. Results show mechanically alloyed V-4Cr-4Ti alloy with Ti₃SiC₂ addition exhibits promising strength at both room temperature and elevated temperatures. Especially, its steady creep rate is almost one order lower than melted V-4Cr-4Ti alloy. The mechanism is considered as the thermal stable nano-particles resisted dislocation motion at high temperature, and is worth being introduced to the strengthening of other structural materials.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 87 / 178

The effect of He nanobubble on inhibiting D trapping in radiation damaged tungsten

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Previous study results show that He nanobubble has great effectiveness on reducing deuterium (D) retention by acting as D diffusion barrier in undamaged commercial ITER grade tungsten (W), however, this effect on radiation damaged tungsten has not been extensively studied. In this paper, He plasma exposure (ion flux: $1023 \text{ m}^{-2} \text{ s}^{-1}$, fluence: 1025 m^{-2}) pre-treatment was performed to create a thin He nanobubble layer in tungsten at $\sim 773 \text{ K}$, after which these samples were irradiated by 5 MeV Cu ions or 5 MeV C ions to induce average peak dpa of 0.001 to 0.1 at room temperature. Samples without He plasma exposure pre-treatment were irradiated by Cu or C ions at the same time. All the samples above were subsequently exposed to D plasma at 350 K to a fluence of 1026 m^{-2} . Nuclear reaction analysis (NRA) was used to evaluate the D distribution profile in the near surface, while the total D retention was measured by thermal desorption spectroscopy (TDS). Possible mechanisms are proposed to interpret the experiment results.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 39 / 179

Design and Fabrication Process of Toroidal Field Coil for HL-2M

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HL-2M is a medium-sized copper-conductor tokamak under construction at the Southwestern Institute of Physics (SWIP). The designed plasma parameters are as follows, plasma current=3 MA, toroidal field = 3 T, major radius =1.78 m, minor radius = 0.65 m, flux-swing > 14 Vs, plasma pulse ~ 5 s, with a plasma shape of elongation = 2 and triangularity > 0.5. The demountable toroidal field coils structure has been selected in order to removal of the vacuum vessel and poloidal coils integrally. The toroidal field coil is designed for a maximum field of 3 T at $R = 1.78 \text{ m}$ and consists of 140 turns (20 bundles, each bundle 7 turns) with a maximum current of 191 kA. Under normal operation condition, the current is 140 kA and the field is 2.2 T. Each turn is composed of inner, upper and outer arc part which are connected by finger joint and bolted joint. The report introduces the structure of toroidal field coils, selection and fabrication process of conductor materials for toroidal field coils, the parameters of conductor materials such as the hardness, electrical conductivity, yield strength, tensile strength, and fatigue strength firstly. Then introduce fabrication technology for toroidal field coils, such as welding cooling water pipe technique, turn to turn insulation technique, prestressed epoxy-glass cylinder technology.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 68 / 180

Establish full covering liquid metal film flows under poor wettability conditions for liquid divertor of fusion reactor

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Investigation on effects of wettability on liquid metal free surface film flow states have been performed by numerical simulations and experiments to establish a film flow which can cover the whole bottom solid surface. The effects of density of fluid, inlet film thickness and the width of bottom solid surface on the film flow states under poor wettability conditions have been investigated by numerical simulations, the results show that the rivulet flow is easily developed when the initial film thickness is small; it is more easily developed to rivulet flow when the fluid density becomes smaller; the covering bottom surface becomes big with the increase of the bottom surface width. But for liquid lithium it is difficult to get the film flow which can cover the whole bottom solid surface by increasing the bottom surface width and inlet film thickness. A new method by using a multi-curve bottom surface has been proposed to solve above problem. Firstly an experiment of the film of GaInSn alloy flow through a chute with a multi-curve bottom surface has been done to validate above solving method, it is indicated that this method is effective and the experimental results are in agreement with numerical results. Secondly numerical simulations have been performed to get the lithium film flows which can cover the whole bottom surface, it is shown that a full covering lithium film flow can be obtained by optimizing the shape of above mentioned multi-curve bottom surface. Above results is valuable for the design of liquid divertor of magnetic fusion reactor.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 112 / 181

Sensitivity Studies of Tritium Transport to WCSB of CFETR

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Primary concept design of CFETR (Chinese Fusion Engineering Test Reactor) has finished. In the fusion reactor, tritium is bred by lithium and then be extracted. However, tritium will contaminate the reactor structures and be leaked out to the environment eventually. For realizing the environmental friendly and tritium self-consistency, a mathematical-physical model is established to analyze tritium transport in CFETR's WCSB (Water Cooled Solid Blanket). Some sensitivity studies of parameters on tritium losses and inventories were performed, the results show that the thickness of tungsten covering the first wall, the hydrogen concentration in coolant water and the water concentration in purge gas have significant impacts on tritium transport process.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 51 / 182

Study on the effect of Pressure on the Electrical insulation of Plasma-sprayed Alumina Coating

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In this paper, 316LN austenitic stainless steel coated with plasma-sprayed alumina for low-voltage electrical insulation is reported, which used in ITER Magnent Support to prevent eddy current loops. The coating will endure the large pressure and experience the several thermal shocks under practical working environment. SEM shows that coating is dense and good adhesions with base materials, and the surface roughness and texture are uniform. After 50 times thermal shocks, the coating is free from cracks, flakes and debonding. The thermal shock has a little influence on the electrical resistance properties. In addition, though the surface contact resistivity decreases with the increase of pressure, it still more than $109\Omega\cdot\text{m}$ when the pressure increased to 250MPa, which can be concluded that the coating is good electrical insulation enough and can be utilized in the Magnent Support of ITER.

Eligible for student paper award?:

No

T.OA2: Divertors and PFCs: Tungsten / 183

Tungsten monoblock concepts for the U.S. Fusion Nuclear Science Facility (FNSF) first wall and divertor

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Next-step fusion nuclear devices require plasma-facing components that can survive a much higher neutron dose than ITER, and in many design concepts also require higher operating temperatures, higher reliability, and materials with more attractive safety and environmental characteristics. In search of first wall concepts that can withstand surface heat fluxes beyond 2 MW/m^2 , we analyzed advanced “monoblock” designs using coolants and materials that offer more attractive long-term performance. These use tungsten armor and heat sinks, similar to previous designs, but replace the coolant with helium and the coolant containment pipe with either low-activation ferritic-martensitic steel or SiC/SiC composite. Two geometries of coolant containment pipe, round pipe and elongated slot (as in microchannels), were examined via 3D thermal and mechanical analysis, which was performed parametrically for optimization. The results of analysis show that helium-cooled steel can remove up to 5 MW/m^2 of steady-state surface heat flux and helium-cooled SiC/SiC can remove nearly 8 MW/m^2 while satisfying all materials and design requirements. This suggests that a He-cooled W/SiC monoblock could withstand divertor-like heat fluxes. More detailed results and conclusions are as follows.

A monoblock with ferritic-martensitic steel round pipes is limited to a steady state surface heat flux of 2.1 MW/m^2 , increasing to only 2.4 MW/m^2 , with the use of advanced steels. The higher allowable temperature of advanced steel can not be fully exploited because in this case the stress limits performance. The use of a slotted “microchannel” geometry provides substantial additional heat flux handling capability. For “ordinary” ferritic steel, the heat flux limit rose to 3.7 MW/m^2 , which roughly meets our original goal to double the performance of the previous He-cooled design with W pins. This value rises to 5.2 MW/m^2 using ODS steel. In this case, stresses did not constrain the performance.

The use of SiC composite pipes to replace steel was considered in the context of large existing R&D programs developing advanced fission fuel cladding. Used inside of a W monoblock configuration,

round pipes can satisfy temperature and stress limits up to $\sim 5 \text{ MW/m}^2$ steady state surface heat flux, whereas a microchannel geometry can reach near 8 MW/m^2 . This value of heat flux approaches the range expected in a tokamak divertor. Exact specifications of heat flux in the divertor of burning plasma devices are not available, but peak values in the range of $5\text{-}15 \text{ MW/m}^2$ are expected.

Our SiC/SiC design variant provides a possible alternative to the He-cooled W-alloy divertor that has been explored in several design studies and R&D programs. The W-alloy divertor has been shown to allow very high performance and heat removal capability, but the availability of an acceptable alloy remains a major uncertainty for its continued development. While SiC composites at present do not achieve the higher heat flux capability of W-alloy, due to limited thermal conductivity, their commercial availability and existing database under neutron irradiation make them a more likely candidate for near-term applications in next-step devices.

The authors would like to acknowledge the contribution of FNSF research group.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 79 / 184

Particle model of the driver of ITER NBI system

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The Neutral Beam Injection (NBI) heating system of ITER tokamak is a key stage for the yield of the full fusion machine. The NBI yield in turn strongly depends on the performance of the first component of the system, the negative ion source of D⁻ ions.

This negative source starts with a “driver region” where a RF discharge is induced in the deuterium gas and a plasma is created. The sources of this kind are therefore referred as Inductively Coupled Plasma –Radio Frequency (ICP-RF) sources. The formed plasma expands in a larger chamber and is then extracted.

While several simulation tools have been developed for the expansion and extraction regions, a full simulation of the driver region is still lacking, mainly because of the difficulties created by the self-consistent inclusion in the codes of the inductive coupling of the RF frequency with the plasma.

For this reason we developed a 2.5D Particle-In-Cell Monte-Carlo-Collision (PIC-MCC) model of a cylindrically symmetrical ICP-RF source, keeping the grid spacing and time step of the simulation small enough to respectively resolve the Debye length and the plasma frequency scales. We report the results of these simulations, which require a massive parallelization, and give some details about the computational side. This is a first step for the modelling of the full negative ion source.

Eligible for student paper award?:

No

W.OP1: Magnets / 185

Progress and Study on the Superconducting Magnet System of China Fusion Engineering Test Reactor

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CFETR (China Fusion Engineering Test Reactor) concept design work was started in 2012. It is developed in two stage. CFETR-Phase I is designed with major radius $R=5.7\text{m}$, minor radius $a=1.6\text{m}$ and magnetic field at plasma region $BT=4-5\text{T}$. 16 toroidal field coils and 6 central solenoid coils were designed using Nb3Sn CICC conductor with maximum operation current of 64 kA and 50 kA, respectively. Three types of plasma equilibrium shapes are designed, namely ITER-like single null, super-X and snowflake. The maximum flux provided by central solenoid is designed as 180 volt-second. However, in order to study high-performance issues such as steady-state particle and heat exhaust, disruption mitigation and avoidance, ELM control, and material damage by high heat flux and neutron, the superconducting magnet system of CFETR-phase II has been updated based on a larger machine with $R = 6.7\text{m}$, $a=2.0\text{m}$, and $BT= 6-7\text{T}$. With this new design, over 1GW fusion power can be achieved and advanced plasma performance can be obtained.

In consideration of the maximum magnetic field of TF coils of CFETR-phase II, a high performance Nb3Sn CICC magnet was designed which can withstand 14-15 T. In order to save the space for blanket system and increase Ohmic heating flux, a high temperature superconducting Bi-2212 magnet with better current carrying performance under high field is considered for the central solenoid (CS) coils of CFETR-phase II. The HTS CS coils can provide a about 480V•s volt-seconds and the maximum magnetic field is about 17.5T. In addition, a Bi-2212 CICC conductor sample was tested at 4.2 K with critical current of 26.6 kA under its self-field.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 25 / 186

Recent improvement of the design of the ITER steady-state magnetic sensors

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The outer vessel steady-state sensors (OVSS), a subsystem of the ITER magnetic diagnostics, will contribute to the measurement of the plasma current, plasma-wall clearance, and local perturbations of the magnetic flux surfaces near the wall. The diagnostic consists of a poloidal array of sixty sensors welded to the vacuum vessel outer surface. Each OVSS contains a pair of bismuth Hall sensors with the measurement axes parallel and normal to the vacuum vessel surface. The OVSS measurement accuracy relies on a precise temperature measurement at the Hall sensors location. This measurement is performed by an on-board thermocouple inserted between the two Hall sensors. The paper describes the recent improvement of the OVSS design to achieve high precision (within 0.1 °C) measurement of the Hall sensor temperature. An explosion-bonded copper plate has been incorporated

in the sensor housing to ensure temperature homogeneity between the Hall sensor and the thermocouple. An indium capsule with the volume of about 0.5 cm³ has been introduced around the thermocouple to allow for in-situ recalibration of the thermocouple during each ITER baking cycle. The design modification has been validated both in thermal simulations and in experiment.

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

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 102 / 187

Modeling and analysis on the six-phase generator - converter system as the magnetic field power supply of HL-2A/M

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The flywheel six-phase generator operated in pulse mode supply power of the magnetic field on HL-2M tokamak. The precise and stable control of six-phase generator is essential for obtaining high beta, steady state plasmas in HL-2M.

Concretely for the power supply of the toroidal field coils on HL-2A, it consists of six-phase synchronous generators with their excitation systems, diode rectifiers, and the toroidal field coils. The diode rectifiers connect the terminals of the generators from AC side, and the toroidal field coils from DC side. During a shot of plasma discharge, the energy stored mechanically in the shafting of generator is transferred to the toroidal field coils by firing the exciter of the six-phase synchronous generator. In this case, modeling on the six-phase synchronous generator operated in pulse mode is the one of key issues for the precise and stable control.

The state space realization of the six-phase generator with related exciters is specifically developed by dual DQ transformation. Based on the electromagnetic dynamics which describes the flux linkage changes in DQ frame, the six-phase generator is represented by 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 speed to decrease from the initial speed, i.e., the energy is released.

The dynamics of the toroidal field current scenario is simulated on the basis of state space realization of six-phase generator built in DQ frame. The results show the consistency with the experimental results of HL-2A, and the realization is effective for the toroidal field current control. Furthermore, this model can be adopted for the preprogrammed feedback control of power supplies on HL-2M. Moreover, it is also suited for the simulation and analysis of synchronous machine - converter systems.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 58 / 188

Electromagnetic Analysis of the ITER Glow Discharge Cleaning Electrode in Equatorial Port No.12

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Glow discharge cleaning (GDC) shall be used on ITER device to reduce and control impurity and hydrogenic fuel out-gassing from in-vessel plasma facing components. After first plasma, permanent electrodes (PEs) will be used to replace Temporary Electrodes (TEs) for subsequent operation. These PEs will be used inside vacuum vessel and during plasma operation, major disruption and vertical displacement events may cause huge electromagnetic forces on these PEs and destroy them. Especially for equatorial ports, the PE is closer to plasma, so analysis has to be done to understand the specification of these EM loads, and these loads will be used as input for structural analysis. This report presents results of transient electromagnetic analysis of electrode structural components during plasma disruption for the seven cases. The calculation is based on the ITER Global Model (IGM) for the EM analysis of vessel components. The method is inserting the model of GDC into the IGM, which provides a certain region for the GDC insertion. The eddy current distribution and the time-varying Lorentz force and torque moment acting on the GDC electrode are presented. According to the calculation results of 7 cases, the maximum moment is about 3.3kNm. From the results the severe case(s) can be determined and only this severe case will be considered if a structure design is updated and this will save lots of resources avoiding consider 7 cases again. Below is a picture showing the moments and forces under 7 different cases.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 80 / 189

Polarizer designed for the electron cyclotron resonance heating system on J-TEXT

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A polarization-controlled launcher has been designed for the 60 GHz electron cyclotron resonance heating system on J-TEXT. The polarizer is an essential component of the polarization-controlled launcher which is used to change the polarization of the electron cyclotron wave. The coordinate transformation method is applied for design of the polarizer with sinusoidal grooves. The polarizer was tested with the lower power test platform. The results agree well with the numerical results of the coordinate transformation method, indicating that the designed polarizer can meet the requirements of the electron cyclotron resonance heating system of the J-TEXT.

Keywords: J-TEXT, ECRH, polarizer, C-method.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 59 / 190

Multi-scenario evaluation and electromagnetic loads on CFETR VV mockup during MD event

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China Fusion Engineering Test Reactor (CFETR), as a new tokamak device to bridge the gap between ITER and DEMO, is developed for further research fusion power plant by China National Integration design Group for Magnetic Confinement Fusion. As a key component to maintain the reliability in run of high-temperature plasma in tokamak, vacuum vessel (VV) has a direct influence on the operation security of the total device. In order to establish the fabrication technology of VV, CFETR vacuum vessel mockup is constructed by Institute of Plasma Physics Chinese Academy of Sciences (ASIPP), its design parameters come from the China Fusion Engineering Test Reactor (CFETR). CFETR magnet system is required to meet the requirement of three scenarios of coil currents, which are used to realize the ITER-like, snowflake and Super-X plasma equilibrium shapes, respectively. In this paper numerical analysis is performed for the electromagnetic loads on CFETR VV mockup corresponding to three different current scenarios shapes during the MD event, respectively. The finite element model for electromagnetic analysis include a 22.5° VV sector and a magnetic system including 2 halves toroidal field (TF) coils, 6 poloidal field (PF) coils, 6 central solenoid (CS) coils, 2 divertor coils (DC) has been built, and a detailed CFETR VV mockup finite element model is established which consists of inner shell, outer shell, reinforcing ribs, ports and magnet coils, etc. The current loads are applied by current density method. The influence of plasma equilibrium configuration on the eddy current and electromagnetic force is also analyzed. The electromagnetic loads on VV during major disruption (MD) will provide a technical support for the future structural design and loads evaluations of CFETR VV.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 113 / 191

Tritium transport analysis for one water-cooled ceramic breeder blanket module of CFETR based on COMSOL

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Tungsten and RAFM steel are respectively used as armor material of first wall (FW) and structure material. The Li₂TiO₃/Be₁₂Ti mixed pebble bed is selected to fulfill tritium breeding for water-cooled ceramic blanket (WCCB) in China Fusion Engineering Test Reactor (CFETR). Tritium would be existed in all materials of the blanket because of its diffusivity and permeability. Thus, it is crucial to study the tritium retention and permeation in the main domains for the safe operation of blanket, especially considering the radioactivity of tritium. Based on the Finite Element Method (FEM), a two-dimensional tritium transport model is set up using COMSOL Multiphysics. The temperature field is constructed to display the temperature distribution within the model, considering the nuclear heat generated and the cooling of the coolant. The velocity field of He purge gas is simulated to analyze the ability of the helium purge gas making tritium entering into Tritium Extraction System

(TES). Meanwhile, the processes of tritium diffused through the interface of different materials, permeated into the coolant and taken by purge gas are considered to calculate the tritium retention and permeation coupling with the temperature and velocity fields. The calculation results indicate that the permeation amount into the coolant of FW is 2.18×10^{-6} mg/s, while that of four cooling plates is 2.70×10^{-6} mg/s. In the purge gas, the tritium would be carried out at the speed of 1.27×10^{-3} mg/s, assuming the existing HTO totally entering into TES. The retention in the inner pebbles is 0.03g. The accuracy of the result is also discussed in this paper.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 2 / 192

Pebble Bed Thermo-mechanical Modeling for Water Cooled Ceramic Breeder Blanket for CFETR

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The beryllium pebble bed and Li₂TiO₃/Be₁₂Ti mixed pebble bed are selected to realize neutron multiplication and tritium breeding in the Water-cooled ceramic breeder blanket (WCCB) of China Fusion Engineering Test Reactor (CFETR). In order to evaluate and improve the performance of WCCB, studies of the thermo-mechanics of the concerned pebble beds are necessary.

In the current research, a numerical model was constructed by using distinct element method (DEM) to analyze behavior of prototypical blanket pebble bed. A thermal contact model based on SZB model was developed to analyze heat transfer in pebble bed. Besides, a numerical analysis program for uniaxial compression test was performed to estimate the macro-meso mechanical behaviors of pebble beds. The suitability and validity of the current numerical model were evaluated by comparing with the previous experimental or numerical data. According to the current calculations, the results of both effective thermal conductivity estimation and pebble bed loading/unloading analysis agree well with the previous experiments.

Finally, the model was extended to the pebble bed analysis of WCCB. A series of numerical simulation work, including steady-state thermal analysis, uniaxial compression test were conducted to obtain basic pebble bed characteristic parameters, such as effective thermal conductivity and strain-stress relation. This study will be dedicated to present the heat transfer features, macro-meso mechanical behaviors and the thermo-mechanical coupling characteristics of the blanket pebble beds, especially the Li₂TiO₃/Be₁₂Ti mixed pebble bed for WCCB.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 3 / 193

Thermal hydraulic analysis for one water cooled blanket module of CFETR based on RELAP5

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The Water Cooled Ceramic Breeder blanket (WCCB) is one of the blanket candidates for Chinese Fusion Engineering Test Reactor (CFETR). The conceptual of WCCB for CFETR under 200MW fusion power has been designed based on pressurized water cooled reactor (PWR) technology. RELAP5 code, which is mature and often used in transient thermal hydraulic analysis in PWR reactor, is selected as the simulation tool. In this paper, the nodal model for RELAP5 is developed corresponding to typical WCCB module, i.e. the coolant passages inside the module were nodalized as hydrodynamic components and the associated module components were simulated with one dimensional heat structures. The steady state characteristic under full power is analyzed. The stable fluid and wall temperature distributions and pressure drops are studied. The results are agree with those of the two dimensional CFD analysis by FLUNET. Furthermore, the transient characteristics under three accidental scenarios, in-vessel loss of coolant accident (LOCA), in-box LOCA and ex-vessel LOCA, are analyzed, respectively. Simulation results show all the temperature of structure wall are within the design limitation and the decay heat can be removed by radiation heat transfer in the three LOCA scenarios, also the pressure of the related volume is within the limits.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 73 / 194

ESTIMATION OF STRAY CAPACITANCES OF TWIN SOURCE HVDC TRANSMISSION LINE AND ITS STORED ENERGY

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Abstract:

In Neutral Beam Injector (NBI), ions are accelerated to desired energy (from ~10 kV to ~MV range) by an electrostatic multi-aperture grid system of an ion source. Accelerated ions subsequently neutralized in a gas cell called neutralizer. To maintain the electrostatic lens configuration, grid plates are placed closely packed (~mm distance) parallel to each other. Paschen-breakdown (here it is called as grid breakdown) between the grid plates occurs routinely during system conditioning phases due to the presence of high voltage (HV) and sufficient gas (@ sub-atmospheric pressure). When grid-breakdown occur the stored energy of HVDC transmission line is dumped into the grids of the ion source at the breakdown location and possesses a danger to damage the grids by melting and even puncturing the spot.

In Twin Source 1 120meter long transmission[2] line is designed to connect accelerator power supply system -35kV ,15 A and extraction power supply system -11kV ,35A with Plasma Grid(PG) , Extraction Grid (EG) and Ground Grid(GG) .The major contribution for the stored energy emanates from the inter conductor capacitances or stray capacitances of the HVDC transmission line. This paper discusses the methodologies for estimation of the inter conductor capacitance and thus stored energy. The exercise helped to get the optimized possible transmission line configuration to ensure

low stored energy to avoid grid damage. The analytical calculations of aforesaid configuration is validated with simulation, performed on COMSOL platform and the corresponding obtained results is further confirmed from capacitance measurement of 1m long prototype HVDC transmission line in similar configuration. The stray capacitances for 1m long HVDC transmission line between PG Line, EG Line and GG Line i.e. CPG EG, CEG GG and CPG GG estimated analytically are 13.41pF, 19.67pF and 8.6pF respectively. The simulated values are 19.41 pF, 18.45pF and 10.46pF respectively. The measured values are 16.86pF, 15.52pF and 9.67pF respectively. The estimated stored energy is 12.85mJ for 1m length of HVDC transmission line.

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Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 106 / 195

Automatic Deployment of a Nuclear Fusion Experimental Data Storage Cluster

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With the rapid growing of experiment data, widely using of distributed database or file system will be the trend in future fusion storage. For such storage systems, a cluster is required, but its setting up and service's deploying are challenging. Besides, plenty of configurations should be set carefully, including environment variable and software runtime, which is a complicated task. Apart from that, due to the variety of hardwares and difference between test and production environment, particular setup procedures are required. J-TEXT Cloud Database (JCDB) is a nuclear fusion experimental data storage and management system, aiming to satisfy the requirements of future long pulse experiment. Based on MongoDB server, Cassandra cluster servers and web applications servers, it will cost much time for JCDB to setup these servers from the beginning. To scale out and configure more conveniently and automatically, JCDB used Docker, a lightweight software containerization platform, to build cloud database and web applications, guaranteeing consistency of the all servers and reducing plenty of time. JCDB pushed two images to Docker repository and scaled out by pulled specified images. Even without any servers on hand, you can also deploy it on cloud services such as Amazon Web Services, Microsoft Azure.

Eligible for student paper award?:

Yes

W.OA2: Divertors and PFCs: Liquid Metals / 196

Investigation of an upgraded flowing liquid lithium limiter for higher performance plasmas in tungsten divertor in EAST

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Flowing liquid Li (FLiLi) used as plasma facing components (PFCs) promises to improve plasma performance due to reduced fuel recycling and impurity generation¹. Various static and flowing liquid Li limiter have been tested in HT-7 and EAST, and significant progress has been achieved from the first FLiLi limiter based on the concept of a thin flowing film in the EAST device in 2014^[2]. First it was confirmed that liquid Li could be driven by an innovative in-vessel DC EM pump to form a re-circulation loop. In addition exciting new results were also obtained during the FLiLi operation, including a controllable Li emission layer at the plasma edge due to the strong interaction between the liquid Li surface and plasma. This effectively reduced particle and heat flux onto the divertor plate, and mitigated ELMs with evidence of short (<150ms) ELM-free phases. However, it also encountered some issues, including clogged distributor before FLiLi experiment, non-uniformity Li distribution on limiter plate, and damaged limiter surface.

In order to enhance Li coverage uniformity and erosion resistance of limiter surface, two electromagnetic pumps were used to drive liquid Li on a copper plate with a stainless steel protection barrier. Hot isotatic pressing (HIP) technology was applied to improve the combination between SS thin layer and Cu heat sink, and the thickness of the SS surface layer was increased from 0.1mm to 0.5mm. Also, electric spark and successive wire-cutting was used to obtain a more uniform distributor design for improved Li flow uniformity. A set of high pressure He cooling system was designed to control the limiter temperature. By upgrading the FLiLi during 2016 FLiLi experiments, significant engineering improvements were demonstrated, resulting in improvement of liquid Li coverage uniformity >80%, as compared to about 30% in 2014 FLiLi campaign. In addition the present limiter surface was undamaged by PMI, in contrast to the 2014 results. It was also confirmed that high pressure (~3MPa) He was more effective to cool limiter than N₂ and Ar during plasma discharge. Moreover, improved plasma performance during full-field ohmic discharge and transient ELM-free H-modes with strong increase of WMHD and H98 were demonstrated for the first time with the new limiter.

These promising results are encouraging for the use of flowing liquid lithium PFCs for future devices.

Reference

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Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 4 / 197

Prediction of departure from nuclear boiling in the first wall of WCCB blanket for CFETR

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The Water Cooled Ceramic Breeder blanket(WCCB), which employs the operating conditions of Pressurized Water Reactor(PWR), namely inlet/outlet temperature of 285/325°C and pressure of 15.5MPa, is being comprehensively researched in the Institute of Plasma and Physics Chinese Academic of Sciences. As an important component of the Water Cooled Ceramic Breeder blanket(WCCB), the first wall faces the plasma directly. It removes away the high heat flux and nuclear volumetric heat by coolant water flowing through the internal cooling passages. The departure from nuclear boiling(DNB) is the typical crisis in the reactor which uses water as coolant, especially for PWR, because it can make the water near heated wall dry out, at which point the local temperature can have an excursion exceeding the limits and the integrity of the structure is damaged. The DNB can easily happen in the first wall(FW) channel for the reason that the enhanced radial transport and edge-localized modes(ELMS) in the fusion reactor can increase the heat flux as high as several MW/m². Therefore, the investigation on the departure from nuclear boiling is necessary.

In this paper, the DNB is numerically analyzed by the CFD approach, which has the capacity of solving the Eulerian two-phase equation with Rensselaer Polytechnic Institute (RPI) wall boiling model. Responding to the excursion of heat flux, the main issue concerned is to determine the ultimate heat flux when the boiling instability happened during the normal operation, indicating the DNB occurs. Furthermore, the influence of different structure design on DNB is also investigated. The FW containing the parallel channels is modeled, in which the velocity of each channel is obtained from the previous thermal hydraulic analyses on the blanket under the normal condition, namely the heat flux of 0.5MW/m². Besides, the detailed flow behavior and distribution of two-phase are also revealed. All these results are beneficial for the further safety operation of fusion reactor.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 104 / 198

A flexible web visualization framework for nuclear fusion experiment data

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As the fusion experiment goes to steady state and more sophisticated diagnoses are developed, the experiment data becomes larger and collaboration between researchers tends to be more frequent. So a well-designed flexible and easy to use web visualization framework is becoming more important.

The new web visualization framework is designed and implemented based on ASP.NET MVC framework. It is part of the JCDB project, which is a database cloud for J-TEXT based on Cassandra. In JCDB data are stored in form of matrix and can be read and written efficiently with cursor and writer. MongoDB is used to store data structure. Models are designed to works with the JCDB backend.

In the controller, we design a RESTful web API which allows users to access and operate data through HTTP after authorized. For GRUD data operation, we provide actions with get, post, put and delete

method. For large data transmission, the stream action and binary serialization can be chosen to reduce the network overhead and improve the performance.

In the view layer, we adopt a modular interface which is flexible and highly user oriented. The tree module can present the whole experiment channel in the lazy loading way and allow users to design their experiment data structure. The visualization modules are responsible for data visualization for different channels. Different visualization modules are chosen automatically for different types of data. And users can save or share the setup anytime they like because the URL for the page keeps in sync with the page content and layout. Furthermore, all the modern browsers in intelligent terminals with different size are supported.

This data visualization framework has been deployed and integrated in LogBook, which is a web system for experiment data management and visualization. The delay is usually small and the user experience is much better than that in traditional data visualization tool used in fusion community. With this web visualization tool, the researchers can visualize and analysis the experiment data wherever the Internet covers, and can save and share their experiment data more easily and efficiently.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 69 / 199

Preliminary Cooling Channel Design and Thermal-hydraulic Analysis of GDC PE in UPP14

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The GDC PE in UPP14 is one of the plasma facing components in the ITER tokamak device. It will get a large number of thermal loads such as radiation and charge-exchange, neutron heating. So it would need the active cooling requirements from Diagnostic Port Plugs(DPP).

The electrode model consists of three parts: the electrode head, the electrode rod and pipes. The first layer channel shape is a rectangle (11mm×15mm) except for the first and the last one due to the irregular head shape. The shape of the other channels is all circular and many channels are designed in parallel way for two reasons: The thermal load on the second half is less than the front part; Reduce the flow resistance.

To obtain the reasonable cooling needs, the preliminary thermal-hydraulic simulation and analysis has been done, which is based on the turbulence model.

The pressure loss in the fluid channel is about 0.452MPa(allowable maximum pressure drop 1.35MPa) during POS. The temperatures of the electrode volume and the electrode temperature peak is 277.5°C, which is less than limit peak stainless steel temperature 450°C. Therefore, all the results meet the thermal design under the flow rate 0.7kg/s, and they could provide some references for the next design optimization, such as pressure drop matching with other components, etc.

Key words: ITER, GDC, thermal-hydraulic

Eligible for student paper award?:

No

W.OP2: Heating and Current Drive / 200**OPTICS AND THERMO-MECHANICAL ANALYSIS OF THE ACCELERATOR FOR THE DEMO NEUTRAL BEAM INJECTOR****Author:** Piero Agostinetti¹**Co-author:** Piergiorgio Sonato¹¹ *Consorzio RFX***Corresponding Authors:** piergiorgio.sonato@igi.cnr.it, piero.agostinetti@igi.cnr.it

DEMO (DEMOstration Fusion Power Plant) is a proposed nuclear fusion power plant that is intended to follow the ITER experimental reactor. The main goal of DEMO will be to demonstrate the possibility to produce electric energy from the fusion reaction. The injection of high energy neutral beams is one of the main tools to heat the plasma up to fusion conditions.

A conceptual design of the Neutral Beam Injector (NBI) for the DEMO fusion reactor, is currently being developed by Consorzio RFX in collaboration with other European research institutes. High injector efficiency and low recirculating plant power, which are fundamental requirements for the success of DEMO, have been taken into special consideration for the DEMO NBI. Moreover, a particular attention has been paid to the issues related to Reliability, Availability, Maintainability and Inspectability (RAMI).

A novel design of the beam source for the DEMO NBI is being developed featuring multiple sub-sources, following a modular design concept, with each sub-source featuring its Radio Frequency driver, capable of increasing the reliability and availability of the DEMO NBI. Copper grids with increasing size of the apertures have been adopted in the accelerator, with three main layouts of the apertures (circular apertures, slotted apertures and frame-like apertures for each sub-source). This design is expected to permit a significant decrease of the stripping losses in the accelerator while maintaining good beam optics, in terms of divergence and deflection.

The conceptual design of the accelerator grids and the related beam optics and thermo-mechanical investigation are presented in this paper. The beam optics calculations, in particular, have been carried out using a fully comprehensive model, able to calculate the magnetic field, the electrostatic field and the trajectory of the negative ions in a self-consistent way.

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

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 74 / 201**Power control system of 4.6GHz Lower hybrid wave for experimental advanced superconducting tokamak****Authors:** Wendong Ma¹; Wang Mao¹; Zege Wu¹; Jianqiang Feng¹; Liu Fukun¹; Jiafang Shan¹¹ *Institute of Plasma Physics, Chinese Academy of Sciences***Corresponding Author:** mwd@ipp.ac.cn

The 6 MW/4.6 GHz lower hybrid current drive (LHCD) system as an effective approach for auxiliary heating and noninductive current drive has been designed and installed with twenty-four 250 KW/4.6 GHz high power klystron amplifiers in the experimental advanced superconducting tokamak (EAST). The power control system of 4.6GHz lower hybrid wave (LPCS) in continuous wave mode has been set up, which can control the lower hybrid power and protect the LHCD system. In this paper, the system architecture and software of the LHPCS are presented. The LPCS of 4.6GHz LHCD included the microwave pre-amplifier system, directional coupler, high reflected power protection subsystem, data acquisition and power control subsystem. The microwave pre-amplifier system contains master oscillator box, two power dividers box and 24 pre-amplifier boxes. There were two sets of high reflected power protection systems. They were installed to make sure klystron, ceramic window and other devices in safety once high reflection occurs. Data acquisition and power control computer is set up on the basis of national instruments CompactRIO and PXI system. The software for high reflected power protection subsystem, data acquisition and power control subsystem were based on LabView. Moreover, the experiment of measurement of incident power, High reflected power protection were described here in detail. Finally, High power CW operation and power modulation experimental with feedback controlled low power microwave source to results in EAST were shown here in detail.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 40 / 202

Structural design and analysis of the feeder in the CFETR CS model coil

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According to the CFETR CS model coil properties test requirements, a large cryogenic test platform will be set up. The feeder, as one of the important components of the platform, should be studied to afford the much valued help for the basic test facilities. The paper focuses on the structural design (location, conductor type, insulation, supports and so on) and FE analysis based on the gravity load, heat load and induction electromagnetic force at room temperature/low temperature.

Eligible for student paper award?:

No

R.OP2: PMI and Plasma Edge Physics / 203

DESIGN, CONSTRUCTION AND INSTALLATION OF LIMITER & DIVERTOR OF ADITYA-U TOKAMAK

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Co-authors: Kulav Rathod¹; Joydeep Ghosh¹; Deepti Sharma¹; Madanlal Kalal¹; Y. C. Saxena¹; Kumarpalsinh Jadeja¹; Jayrajsinh Parmar¹; R. Srinivasan¹; Sharvil Patel¹; Rakesh Tanna¹; Nilesh Patel¹; S. B. Bhatt¹

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The ADITYA tokamak ($R_0 = 75$ cm, $a = 25$ cm)¹ having a limiter configuration has been upgraded to a state-of-art ADITYA-U tokamak^[2] with divertor configuration to support the future Indian Fusion program. Limiter and Divertor are the most important subsystems of any tokamak. They are used to form the plasma boundary inside the tokamak and restrict the high-temperature plasma from hitting the vacuum vessel wall and provide protection to in-vessel components. A much better configuration in terms of energy and particle exhaust can be achieved in the divertor configuration, where the outermost magnetic field flux lines are opened up to make them strike on a chosen divertor target. The positions of the limiter and divertor plate locations inside ADITYA-U have been determined based on numerical simulation of the plasma equilibrium profile. The new machine accommodates three different configuration of limiter and divertor assemblies. ADITYA-U is having a toroidally continuous inner limiter with a poloidal extent of $\frac{1}{2}$ of poloidal periphery of vessel. There are two outer limiter assemblies installed at two different toroidal location with poloidal extent of $\frac{1}{4}$ of poloidal periphery of vessel. The divertor plates are toroidally continuous structures located at upper and bottom halves of the vessel. In addition, one pair of the safety limiter which is a poloidal ring of graphite tiles placed inside vessel (at toroidal) symmetrical locations. Initially graphite will be used as plasma facing material (PFM) in all the limiter and divertor plates. Shaped graphite tiles have been fixed on specially designed support structures made out of SS-304L inside the torus shaped vacuum vessel. The dimensions of the limiter and divertor tiles are decided based on their installation inside the vacuum vessel as well as on the total plasma heat loads falling on them. Depending upon the heat loads; the thickness of graphite tiles for limiter and divertor plates is decided. All Limiter assemblies of ADITYA-U have been installed inside vacuum vessel. As the vessel dimensions of ADITYA-U are not suitable for any human to go inside the vessel, installation of limiter tiles along with integration of other in-vessel components on the high field wall side was very challenging. Successful plasma operation with these limiters has been obtained during the first phase of machine operation. The divertor plates will be installed during phase-2 machine operation. In this paper, ADITYA-U limiter and divertor conceptual design, fabrication and installation along with challenges faced will be presented.

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Eligible for student paper award?:

No

T.OA2: Divertors and PFCs: Tungsten - Board: 114 / 204

Defect production and deuterium bulk retention in quasi-homogeneously damaged tungsten

Authors: Feng Liu¹; Guang-Nan Luo¹; Chonghong Zhang²; Yin Song^{None}; Mingzhong Zhao^{None}

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Tungsten (W) is foreseen as the leading plasma facing material (PFM) for future fusion reactors due to its advantageous thermal mechanical properties and relatively low solubility of tritium (T). W-PFM in fusion reactors will experience intense radiation by 14 MeV-peaked neutrons (n), which have long mean free paths on the order of centimeters in solids. T retention in W may greatly increase owing

to the T trapping effects of defects created by neutrons throughout the W bulk. Therefore, T bulk retention in n-irradiated W becomes a significant safety concern. Recently, heavy ions are widely used as surrogates for neutrons to investigate the influence of n-produced defects on T retention. However, the damaged layer of heavy ions is usually limited to a few micrometers beneath the specimen surface and the damage profile is strongly peaked. Hence the effects of homogeneously distributed traps on T retention in W have not been fully understood. In this study, by using ultra-high energy ions and special sample irradiation techniques, we produced a quasi-homogeneous distribution of defects in bulk W; then the deuterium (D, a surrogate of T) retention mechanisms in the damaged W are investigated.

The high-energy heavy-ion irradiation was performed at Heavy-ion Research Facility in Lanzhou. Annealed W foils were irradiated with 122 MeV 20Ne ions in a terminal chamber where an energy degrader for defect distribution tailoring was used. SRIM calculation showed that a quasi-homogeneous distribution of atomic displacement damage to 0.16 dpa within a depth of 50 μm was produced in W. Then results from positron annihilation lifetime characterizing exhibited an extra long positron lifetime component of ~ 400 ps in the irradiated W, indicating the formation of large vacancy clusters. After that, the sample was exposed to D₂ gas at 773 K. Thermal desorption spectra featured a high D release peak at ~ 1010 K and a broad D desorption temperature range (730-1173 K) for the irradiated W, which was very different from the non-tailored W sample (much narrow desorption window). Further transmission electron microscopy characterizing (under and over-focus pairs) showed a large amount of voids with a diameter of ~ 1 nm in the irradiated W. These voids could be the large vacancy clusters formed by heavy ion irradiation and should be the main reason to the high temperature desorption of D. However, whether the broad D desorption window is related to the quasi-homogeneous distribution of voids should be further studied.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 75 / 205

Degradation of Neutral Beam heating & current drive by Alfvénic instabilities

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Neutral beam injection in tokamaks results in a population of energetic particles (EP) that can drive instabilities in the Alfvén frequency range. In turn, instabilities can lead to redistribution or loss of EPs, thus affecting the controllability and predictability of quantities such as neutral beam (NB) current drive efficiency and radial profile of the non-inductive current fraction. In this work, examples from NSTX and NSTX-U discharges featuring robust Alfvénic activity are discussed to investigate the reduction of NB current drive by instabilities. Recent improvements to the tokamak transport code TRANSP enable quantitative, time-dependent simulations of NB-heated plasmas in the presence of EP-driven instabilities. In particular, a new physics-based model has been implemented in TRANSP to account for the resonant interaction between EPs and instabilities, which results in more reliable simulations than previously achieved using a simple, ad-hoc diffusive model. Results show that instabilities can strongly affect the $\int n_{EP}$ distribution function. Modifications with respect to 'classical' EP behavior (i.e., in the absence of instabilities) propagate to macroscopic quantities such as the profiles of NB-driven current and of the local EP power transferred to the thermal plasma species through thermalization. For scenarios with multiple unstable EP-driven instabilities, the computed reduction in NB current drive efficiency can be as high as 40% with respect to classical simulations.

Eligible for student paper award?:

No

T.OA1: Diagnostics and Instrumentation I / 206

Design and Analysis Progress of US ITER Integrated Diagnostic Upper Port 14

Authors: Yuhu Zhai¹; Russell Feder¹; Ankita Jariwala¹; Mark Smith¹; Wenping Wang¹; Jingping Chen¹; Tim Edgemon¹; Han Zhang¹; Guirao Julio²; Silvia Iglesias³; Victor Udintsev²

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ITER is the world's largest fusion device currently under construction in the South of France with over 50 diagnostic systems to be installed inside the port plugs (PPs), the interspace or the port cell region of various diagnostic ports. The Diagnostic First Wall (DFW) and Diagnostic Shielding Modules (DSM) are designed to protect front-end diagnostics from plasma neutron and radiation while providing apertures for diagnostic viewing access to the plasma. Four tenant diagnostic systems will be integrated into the upper port plug 14. The upper visible/IR wide angle viewing system (Vis-IR/Upper Cameras), or WAV system, is installed to provide visible and IR viewing of the inner vessel for machine component protection during plasma operations. The disruption mitigation system (DMS) is installed to mitigate the negative effects of plasma events due to sudden loss of plasma current or control by rapid injection of cryogenic pellets to mitigate the dissipation of the plasma thermal energy, the control of the plasma current quench, and the suppression of the generation of Runaway electrons. The Glow Discharge Cleaning (GDC) system is installed for reducing impurity and provides control of hydrogenic fuel out-gassing from plasma-facing components. The Plasma Position Reflectometry (PPR), for real time determination of the wall-plasma distance, is the ex-vessel tenant installed in the U14 Interspace region.

The PP engineering design and multi-physics analysis has been performed following ITER upper port integration requirements including weight, neutron shielding (100 uSv/hr total dose limit), cooling layout, allowable deflections and structural integrity validation under single and combined load cases. Various DSM design configurations have been analyzed and resultant component integration and mass distribution is optimized to limit its impact to the DFW (IO scope) and in-port diagnostics, to mitigate significant impact from the undesirable VDE (Vertical Displacement Event) inertial loads. The DSM design maintains EM load distribution similar to that from a generic box-like shielding structure, still provides needed stiffness for the protection of on-board diagnostics and structural integrity. To moderate impact from inertial loads due to the Vacuum Vessel (VV) movements during asymmetric plasma VDEs, the rigid lock-in DSM interface was implemented into the U14 port integration analysis models for design validation. Structural integrity of U14 assembly is largely driven by the electromagnetic loads induced on the metallic structural components during plasma VDEs. The in-port diagnostics and the mounting supports, on the other hand, are largely driven by the steady-state thermal loads from volumetric nuclear heating, and the dynamic response of components attached to the DSM-PPS assembly under the VDE inertial loads due to the vessel movements. Progress on the U14 integrated design and analysis is reported. The tenant interface load transfer is also presented in details for in-port system attached to the DSMs as part of the design and analysis tasks for ITER PP engineering.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 41 / 207

Electrical and Magnetic Analyses and Design of New NSTX-U PF1A Coil

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Abstract –The PF1A coil is one of the Poloidal Field (PF) shaping coils on the NSTX-U machine. It is critical for shaping highly elongated, and high triangularity plasmas current. In July 2016, the failure of the PF1A Upper coil resulted in shutdown of the NSTX-U experiment. As part of the causal analysis, it was discovered that several passive structures around the PF1A coil had an adverse effect on the electrical and magnetic behavior of the coil system under AC conditions, more than was expected. This effect, although not a direct cause of the failure, significantly increased the harmonic ripple in the coil current as well as the plasma current beyond the design target, and also caused problems with the magnetic diagnostics. Therefore, several analyses were conducted to understand the electrical and magnetic behavior of the coil system at under AC conditions and to account for it in the new design.

A finite element analysis was first performed to map the magnetic field around the coil and capture the eddy current and magnetic coupling effect of the surrounding passive structures on the effective coil resistance and inductance over a range of AC frequencies. The calculated AC impedance is first compared to field measurement, and then the resistance and inductance, at the characteristic power supply frequency, is put into a detailed electrical model, which includes a detailed representation of the power supply system and electrical network, to simulate the electrical behavior during operation. Results from the electrical simulation were then compared to operational records for verification. Using results from these analyses, the new PF1A coils have been designed, which includes an external reactor to account for the passive structure effect. Same methodology can also be applied to design of the other new PF coils.

Key Words: NSTX-U, PF Coil, Magnetic Coupling, Eddy Current, Thyristor based power supply, effective resistance, effective inductance

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 103 / 208

IPSE DIXIT: A User-Friendly Software Tool for the Design and Operation of Tokamak Power Supplies

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The design and the operations of a tokamak often require to assess the feasibility of a desired experimental scenario with the available power supplies (PSs). After the definition of the evolution of the

current waveforms in the supplied conductors (active elements, AEs) and in the plasma, it is necessary to verify if the PS equipment is able to produce them, also taking into account the conductive structures (mainly of the Vacuum Vessel) that are not directly fed by a PS (passive elements, PEs). In an operating tokamak, this process aims to estimate, with a reasonable trade-off between computation time and accuracy, the achievable experiments, the consequent stress on the components and the power demanded from the external grid. In the design of a new facility, this is essential to identify the component ratings and specifications and the impact on the electrical distribution systems.

A user-friendly software tool was developed to answer the question “Is Power Supply Equipment Designed Implementing Current Scenario In Tokamak?” and was named IPSE DIXIT after the acronym of such a question.

The main input data (mutual inductance matrix, resistance vector, currents in the AEs) can be entered with any mesh and time resolution. The effect of the PEs can be obtained by including their inductances and resistances in the input data. A more refined simulation of the plasma and of the AEs can be implemented by modeling them as distributed axisymmetric current filaments. Worst-case or random waveforms can be selected to include the influence of the coils for the feedback control of the plasma position and instabilities.

Some functional models are available for the most common devices and system used in tokamaks, as AC/DC converters, thyristor bridges, switching network units (SNU) and boosters. The current actually flowing in each thyristor of parallel and back-to-back bridge configurations can be estimated, also taking into account the limit firing angles and the circulation currents. The knowledge of the heatsink characteristics allows the calculation of the thyristors' power loss and junction temperature. The verification of the single SNU resistor elements is also possible.

The electrical contribution of the systems for plasma additional heating and of the plant auxiliary services can be estimated by simplified models.

The final result consists in the estimation of the active, reactive and apparent powers expected for each tokamak operation.

IPSE DIXIT was used to design the PSs of the Divertor Tokamak Test (DTT) facility moving from a reference single-null scenario. The results obtained considering only the AEs were compared to those obtained introducing the PEs by meshes of increasing complexity (at least 140 elements). A similar analysis was carried out also on JT-60SA, where the PS components are already defined. In this case, specific thresholds can be set to verify a selected scenario.

The target of the research is to provide a free environment that could be adopted independently of the considered tokamak, covering a wide set of PS topology and components with a possible interaction with existing tools for structural analysis of experimental scenarios.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 105 / 209

A New User Front-End for EAST Remote Participation

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To provide high-efficient and low-cost way to meet international collaboration requirements for the EAST, a web-based remote participation system (RPS), details of which were first reported in the 10th IAEA Meeting in 2015, has been developed. The RPS team focused on the extension, update and optimization for the RPS during last two years. The purpose of this paper is to provide an update of the Remote Participation System in EAST Tokamak. EAST RPS has established Apache-Flex based front-end components to provide a plant-cross user interface since 2012. However, some Web browsers, such as Firefox and Microsoft Edge, disable the flash player plugin by default and the Flex technology will become less relevant in the future. The front-end migration should be a priority to update the EAST RPS. The open source, cross-platform, maintainability and life-cycle are the key features the front-end platform must have. Bootstrap which was provide by twitter, was select to be

the front-end platform for EAST RPS. Bootstrap is a HTML, CSS and JS framework for developing desktop and mobile projects on the web and the solutions are offered in this paper.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 21 / 210

Preliminary Design for Diagnostic Port Integration at ITER Upper Port #18

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ITER has many ports to install various diagnostics which view and measure various plasma parameters. One of the ports, the upper port #18 (UP18) is designed to integrate three tenant diagnostic systems: VUV (Vacuum Ultra Violet) spectrometer, NAS (Neutron Activation System), UVNC (Upper Vertical Neutron Camera). The key design drivers for the port integration are requirements on neutron shielding and maintenance. In this paper, we discuss the neutron shielding design made following the ALARA (as low as reasonable achievable) principle in order to reduce the shut-down dose rate in the interspace and port cell which are human-accessible areas. The design choice for radiation shielding of electronics in the port cell is also discussed. The port maintenance in ITER consists of RH (remote handling) operation for the port plug and manual (or assisted-manual) operation for the interspace and port cell areas. The compatibility with the ITER maintenance strategy is investigated for UP18 and the associated issues are addressed.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 22 / 211

Spectroscopic diagnostics for negative ion source test facility at ASIPP

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In order to support the development of the negative ion based neutral beam injection system for next generation fusion experimental reactor, a negative ion source test facility with radio frequency source is being built at Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP). A full set of spectroscopic diagnostics was designed to meet the requirement of operation performance optimization. This paper describes the design of optical emission spectroscopy of the plasma source and negative ion absorption spectroscopy. Ion formation and recombination processes are accompanied

by radiation of different characteristic spectral lines, the Balmer series and Cs line at 852.1nm. By measuring the intensity of spectral lines, the key parameters, such as temperature and density of electron and cesium quantity, will be estimated. Absorption spectroscopy, with a cavity ring-down technique, will provide a direct measure of the negative ion density. These diagnostics will provide strong support for the coupling of RF power and improvement of negative ion density.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 52 / 212

Effect of Rapid-forging and annealing on the properties of W-TaC alloys

Authors: Fan Feng^{None}; Youyun Lian^{None}

Tungsten materials have been attracting growing interest as a promising candidate for plasma facing materials (PFMs) based on its high melting point, high temperature strength, and good thermal conductivity as well as its low erosion in fusion radiation environment and low tritium retention. But the inherent disadvantages of tungsten materials, including poor low temperature machinability, low ductility, high ductile-brittle transition temperature (DBTT) and irradiation-induced embrittlement, cannot be ignored for fusion reactor application. In recent studies, some process strategies focus on manufacturing nanostructured tungsten materials through thermo-plastic deformation treatment to resist the defect of tungsten.

In this work, the mechanical properties and transient thermal shock performance of W-TaC alloys prepared by hot pressing(HP) followed by rapid-forging and annealing treatment were investigated. Tungsten powder and TaC powder were mixed through High-energy ball milling and then sintered by hot pressing with temperature of 1800°C and pressure of 30MPa. Density, hardness, Tensile strength and total elongation at different temperature of W-TaC alloys were tested respectively. The polished tungsten surfaces were exposed to repetitive ELM-like thermal shock loads at different base temperature and various absorbed energy densities. The thermal shock-induced damages and the microstructure were analysed by scanning electron microscope. The results indicate that strength and ductility can be improved by rapid-forging and subsequent annealing.

Eligible for student paper award?:

Yes

T.OP3: Project Management and Systems Engineering / 213

STATUS ON DESIGN AND CONSTRUCTION OF THE ITER BUILDINGS AND PLANT SYSTEMS

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The Tokamak Complex building construction is progressing on the ITER site. This consists of the Tokamak building hosting the machine, the Tritium building and the Diagnostic building, together with a wide range of auxiliary buildings, such as those for the heating systems, cryoplant, power supply, control of the entire plant or the assembly of the Tokamak machine. The majority of the auxiliary buildings are currently in the execution phase on site.

The ITER project applies a staged approach, with the First Plasma phase scheduled in 2025, followed by the Pre-fusion Power Operations I and II, and finally the Tritium phase scheduled in 2035. As a consequence of the staged approach, the schedule of the building construction and plant systems installation have to be organized accordingly, and the plant configuration for each phase to be defined.

While the manufacture of the long lead items for the Tokamak machine is currently ongoing, the plant systems are completing the final and manufacturing design, in order to start the manufacture of the components, and to be ready on time for the start of the installation works in the buildings.

This paper gives an overview of the building overall design process, and the construction status of each building. The plant layout for the First Plasma is described, with the systems requirements to be fulfilled in order to achieve the first plasma.

This paper further highlights the definition freeze and control, for the physical interfaces, that is needed so as to decouple the plant systems design, reducing the risk of possible changes propagated to the more advanced and mature interfacing systems. The accomplishments and planning towards the Manufacturing Readiness Review for the plant systems, which are required for First Plasma, are summarized.

An overview of the final configuration is presented, to prepare the installation and construction of the plant systems in the buildings, including allowance for the shared use among the contractors.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 91 / 214

Type Tests of JT-60SA Central Solenoid / Equilibrium Field (CS/EF) Super-Conducting Magnet Power Supplies

Authors: Pietro Zito¹; Alessandro Lampasi¹; Luca Novello²; Makoto Matsukawa³; Katsuhiro Shimada³; Shoichi Hatakeyama³; Marco Portesine⁴; Ander Dorronsoro⁵; Dionisio Vian⁵; Koldo Celaya⁵

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JT-60SA is a Superconducting Tokamak in the framework of the Broader Approach Agreement between Europe and Japan. For this International Project among its various procurements, the Italian National Agency for New Technology Energy and Sustainable Economic Development (ENEA) is providing: four AC/DC converters for the central solenoid superconducting magnets (CS1, CS2, CS3 and CS4 PSs rated ± 20 kA and ± 1 kV); two AC/DC converters for the Equilibrium Field superconducting magnets (EF1 and EF6 PSs rated in the range $+10$ kA -20 kA and ± 1 kV); and two AC/DC converters Fast Plasma Position Control Coils (rated ± 5 kA and ± 1 kV). Furthermore, ENEA is also providing six converter transformers of which: four dry type for FPPCC PSs and two oil type for CS PSs. These systems, being procured by ENEA through a contract signed in August 2013 with Industrial Suppliers (Poseico and Jema in Join-Venture), have been already constructed and they are

in advanced testing phase.

The basic devices of a CS/EF PS consist in: a thyristors rectifier or base PS, a converter transformer, a crowbar (to protect by over-voltages and/or over-currents); whereas the load of the CS/EF PS is a Super-Conducting Magnet, whose purpose is the generation of strong magnetic fields able to induce and confine the plasma current in a prefixed geometrical configuration. This requires that the control system of CS/EF PSs is able to reproduce in real-time a reference current or voltage profile, known as "scenario". The scenarios are particularly critical in these systems for the high currents required and hence for the high magnetic energy stored in the load. The main characteristics of CS/EF base thyristor rectifier procured by ENEA are: 4-quadrant AC/DC converter 12-pulses with circulating current in back-to-back configuration, with a DC current of ± 20 kA (+10 kA/-20 kA for EF1 and EF6), a DC voltage of ± 1 kV and an accuracy of $\pm 1\%$. After the design phase, completed in 2015, the construction and the factory tests of all EF/CS PSs were completed in 2016, except EF1 and EF6 PSs, whose tests are currently ongoing.

All performed type tests were carried out both in closed loop feedback current control and in open loop feed forward voltage control and in accordance to the IEC60146 Standards. The electric load used for these tests consists of three inductances connected in series, and the whole resulting values of 3.3mH and 3m Ω represented a worst condition for the current derivative compared to the actual load. The testing facility is fed by the 30kV electric power grid and by an autotransformer, that allows the possibility to increase or decrease the voltage at the primary side of the converter transformers, that power the EF/CS PS. These tests, made at full current, pointed out a good dynamic behaviour of the control system both for forward and for reverse rectifier bridges, assuring the smooth transitions between different operating modes: single mode, circulating current mode and dual mode. This is the focus of the paper. The complete systems will be delivered to Japan by 2017.

Eligible for student paper award?:

No

W.OA3: Neutronics and Multiphysics Analysis / 215

Estimate of Air Activation at the ITER Neutral Beam Test Facility

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In the framework of the safety analyses for the PRIMA (Padua Research on ITER Megavolt Accelerator) ITER Neutral Beam Test Facility (NBTF), activation of the atmosphere in the main experimental hall may represent a major radiation protection concern. The primary goal of this study was to assess radionuclide production in air due to neutron activation. For this purpose, all available input data were considered, including characteristics of the neutron field, air composition, room volume and functional parameters of the ventilation system.

PRIMA comprises two independent test-stands: i) the negative ion source SPIDER (Source for the Production of Ions of Deuterium Extracted from an RF plasma) that will produce hydrogen and deuterium ions and accelerates them up to 100 kV, and ii) MITICA (Megavolt ITER Injector and Concept Advancement) a first full-size and full performance ITER injector, that will accelerate the same ions up to 1 MeV.

The high neutron yield (an average of about $2 \cdot 10^{12}$ n s⁻¹ has been evaluated for MITICA) due to D-D reactions in the components of the NBI experiments, has the potential to produce an important inventory of radionuclides in the air of the accelerator vault. The analysis presented in this work shows that, in this context, the main safety concern is related to the production of ⁴¹Ar during D2 operations of the MITICA facility. The concentration of the ⁴¹Ar in the atmosphere of the main vault has been assessed for the different operational scenarios defined in the design of the PRIMA

experiment. Preliminary results indicate that the highest saturation activity of ^{41}Ar , obtained considering the worst-case scenario, is about 5 GBq. However, due to air changes, the relatively short pulse duration and the ^{41}Ar half-life, saturation activity is never reached inside the vault. Considering the minimum expected ventilation rate, our preliminary analysis indicates that the actual activity of ^{41}Ar stabilizes at about 1.5 GBq. The estimated occupational effective dose rate evaluated immediately after shutdown is about $80\ \mu\text{Sv/h}$, with radiation level dropping in time. On the basis of the data from this study, we conclude that both ventilation rate and waiting time before accessing the experimental hall need to be carefully assessed to prevent undue personnel radiation exposure.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 70 / 216

PRELIMINARY DESIGN FOR THE FIRST WALL IN WEAK MAGNETIC SIDE OF HL-2M PROJECT

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HL-2M tokamak is considered as one of the most important short pulse device for future fusion research in China which is being built at Southwestern Institute of Physics. In the vacuum vessel of the HL-2M, The first wall in weak magnetic side is designed to protect the vacuum vessel, cryopump, RMP coils and diagnostic components from the plasma particles and heat loads. Currently, the preliminary design for the first wall in weak magnetic side is in progress.

Considering the risk of leakage and complexity of design, a passive cooling structure is adopted in the first wall of the weak magnetic side. In order to enhance thermal transfer, the first wall is made of copper alloy (CuCrZr) and graphite tile. Transient thermal analysis has been used to predict heat load for normal operating scenarios in a day. The maximum temperature of this first wall is about 307°C which is engendered on the graphite tile. After a day of operating, the temperature of the passive cooling first wall can be reduced to 54°C .

As a consequence of the high temperature, the stress between graphite tile and copper alloy need more attention. Spring washer and pressure bar have been carried out to optimize the mechanical joint. Flexible copper sheet is placed in joint faces to increase thermal contact resistance.

Eligible for student paper award?:

Yes

M.OA3: Inertial Fusion Engineering and Alternate Concepts / 217

Fusion chamber dynamics and first wall response in a Z-pinch driven fusion-fission hybrid power reactor (Z-FFR)

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The Z-Pinch driven fusion-fission hybrid reactor (Z-FFR) concept utilizes energetic neutrons produced by D-T fusion to drive a sub-critical fission blanket for energy production. Benefiting from an innovative local-holistic-ignition Z-pinch fusion target and advanced sub-critical fission blanket made of natural or depleted uranium, the Z-FFR has significant advantages in safety, economy and environment, and has great potential to be a millennial energy that could address the main issues of long-term sustainability related to nuclear power: fuel supply, energy production, and waste management. In Z-FFR, the fusion target will produce enormous energy of ~1.5 GJ per pulse at a frequency of 0.1 Hz. Almost 20% of the fusion energy yield, approximately 300 MJ, is released in forms of pulsed X-rays. Radiation hydrodynamics in the fusion chamber were investigated by MULTI-1D simulations. To evaluate the influence on thermal and mechanical loads on the first wall brought by the uncertainties of calculated radiation opacities, as well as limitations from employing a single-group treatment in chamber radiation transport, artificial adjustments of opacities by multiplying a coefficient were adopted in the simulations to increase the design reliability. The thermo-mechanical response in a tungsten-coated Zr-alloy first wall was performed by FWDR1D calculations using the derived thermal and mechanical loads as inputs. The temperature and stress fields were analyzed, and the corresponding elastic strains were conducted for primary lifetime estimations. Both pure tungsten and W-Re alloy were tested on an intense pulsed Z-pinch X-ray source to find the safety thresholds of certain materials for designing requirements, as well as to verify and validate the FWDR1D code.

Eligible for student paper award?:

No

W.PLN: Plenary W / 218

Technical Progress of EAST Tokamak

Authors: Yuntao Song¹; Baonian Wan²; Jiangang Li¹; Xiaodong Zhang¹; Peng Fu¹; Xinchao Wu¹; Youwen Sun¹; Junyu Zhao¹; Junling Chen¹; Shoubiao Zhang¹; Peng Zhao¹; Kun Lu¹; Jing Wei¹; Yiyun Huang¹; Liuwei Xu¹; Ge Gao¹; Yu Wu¹; Jingang Qin¹; Huajun Liu¹; Xiang Gao¹; Guoqiang Li¹; Nong Xiang¹; Guosheng Xu¹; Liqun Hu¹; Jiansheng Hu¹; Xianzu Gong¹; Bingjia Xiao¹; Biao Shen¹; Feng Wang¹; Qiyong Zhang¹; Ming Zhuang¹; Hanshen Feng¹; Fukun Liu¹; Yanping Zhao¹; Xinjun Zhang¹; Guangnan Luo¹; Songlin Liu¹; Chundong Hu¹; Yuanlai Xie¹; Bo Lv¹; Qiongqiu Liu¹; Jiefeng Wu¹

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The Experimental Advanced Superconducting Tokamak (EAST) is a superconducting tokamak, which successfully achieved the first plasma discharge in 2006. The major radius of EAST plasma is 1.9m, and its central magnet field is 3.5T. In the past few years, EAST has made many achievements such as 60s double-null divertor configuration plasma and 1MA plasma. In 2016, 102 second long-pulse discharge with plasma center temperature of 5×10^7 degree and 0.4 MA plasma current. The steady-state H-mode plasma more than 60 seconds in the middle of October was also obtained, with further optimization of steady-state operation integrated control under long plus time scales. The H-mode plasma only with RF waves through coupling of LHWH, ECRH and ICRH and complete non-induced current drive was realized, which is an important reference of the operation for ITER; For the engineering aspect, two sets of 4MW neutral beam injection (NBI), one set of electron cyclotron resonance heating (ECRH) and lower hybrid current drive (LHCD) were installed to achieve more than 30MW total heating power. The upper divertor was updated to W-Cu divertor which consists of monoblock structure and tungsten armor that can withstand $10 \text{ MW} \cdot \text{m}^{-2}$ heat load. The high steady-state plasma operating mode of EAST and its experiment work under the condition of tungsten divertor will provide good reference for China Fusion Engineering Test Reactor (CFETR). ITER will adopt RF low-power injection mode and active water-cooled tungsten divertor structure. EAST is one of the best superconducting tokamak in the world with these two features and long pulse operation capability. The stable operation mode will provide important reference for future fusion test reactors.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 63 / 219

Analysis and derivation of the EU-DEMO high level plant requirements**Authors:** Matti Coleman¹; Gianfranco Federici²¹ UKAEA / EUROfusion² EUROfusion**Corresponding Authors:** gianfranco.federici@euro-fusion.org, matti.coleman@ccfe.ac.uk

Ultimately, a DEMO fusion power plant must mature the physics, design, and engineering/technology basis for a future fusion power plant (FPP). In doing so, it must also enable an extrapolable assessment of the economic performance of an FPP. Following discussions with stakeholders and fusion reactor experts, these goals have been further refined into a set of high-level plant requirements. In this work we analyse the implications of these requirements, and derive further requirements which can begin to be allocated to the various plant sub-systems.

In particular, the following requirements are discussed and analysed in more detail:

- Targeted overall plant availability (30%)
- Maximum shutdown duration for maintenance (250 days)
- Tritium self-sufficiency
- Provision of tritium for a fusion reactor beyond DEMO

The implications of the plant availability and unplanned shutdowns on the required tritium breeding ratio and tritium start-up inventory are assessed. The requirement for tritium self-sufficiency necessarily leads to a requirement to provide a tritium stockpile buffer in the event of unforeseen shutdowns. An attempt is made to define a term for this required stockpile of tritium and the implications on the TBR are shown. The presently unconfirmed and ill-defined requirement for DEMO to provide tritium for a future FPP is discussed and attempts are made to reach a coherent and reasonable definition of this requirement. A preliminary assessment of the impact such a requirement would have on the required TBR is made. Sensitivity studies are performed on the maintenance shutdown durations to determine their impact on other high level requirements. Requirements for the overall plant availability are refined and preliminary attempts are made to sub-divide and allocate the availability budget to sub-systems in the form of a lifetime reliability target.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 5 / 220

Design and Analysis of “Filling-Evacuating” High-Pressure Helium-Cooled Loop**Authors:** Haifei Deng¹; Weihua Wang²; Desheng Chen²; Shenghong Huang³; Jinhon Yang⁴

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The breeder blanket and divertor are crucial plasma facing components (PFC) in a fusion reactor. The helium cooled blanket and divertor concepts have exhibited the best potential to come up to the highest safety requirements and therefore been chosen for the development object. As a result of high heat flux radiated from the plasma in the fusion reactor and high power density nuclear heat deposited by high-energy neutrons, the cooling of the First Wall (FW) and the discharge of nuclear heat have become one of the major technical challenges. To demonstrate and verify the helium-cooled technology and tools of China Test Blanket Module (TBM), and explore the feasibility and key technology of thermal hydraulics process of helium-cooled divertor, we have creatively adopted the “filling-evacuating” approach to design and fabricate the High-Pressure Helium-Cooled Loop (HPHCL), in which a mock-up of reduced-scale helium-cooled blanket module is designed and manufactured as a test section. Based on different experimental cases, the operating pressures of helium at mock-up range from 3 to 10 MPa and the maximum mass flow rate can reach up to 0.21Kg/s. In this paper, the design scheme of the HPHCL is presented, and the key issues of engineering manufacture and the test cases are calculated and analyzed. The helium gas flow and the heat transfer are calculated according to the test working conditions of the referenced ITER TBM’s FW surface heat flux, using ANSYS fluid dynamics software FLUENT. The results will provide support for the follow-up fabrication of test system and implementation of the tests.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 88 / 221

Preliminary Assessment of Tungsten as an Optional Plasma Facing Material in CFETR

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Since tungsten is considered as the optional divertor target material for the future fusion device, e.g. CFETR, it is crucial to keep this high-Z impurity concentration under an acceptable level to avoid significant degradation of core performance. In this work, a parameter scan study is performed to preliminarily assess the tungsten impurity. The OEDGE (OSM-EIRENE-DIVIMP) code package is employed, where OSM-EIRENE provides 2D scrape-off layer (SOL) plasma background, and DIVIMP code then simulates the impurity distribution. Instead of specifying the upstream condition, the target plasma parameters are scanned by assuming the heat load of the tungsten divertor lower than the engineering heat flux limit (10MW/m²). A large range of plasma profiles are sampled by the scan of the edge plasma temperature and density decay lengths, which are assigned based on empirical equations. The results reveal both the temperature and density decay lengths have a noteworthy effect on tungsten sputtering flux, divertor tungsten retention and core concentration. The impact of the poloidal drift velocity, radial pinch velocity and cross-field diffusion coefficient on the tungsten transport is also studied.

Eligible for student paper award?:

No

W.OP1: Magnets / 222**Qualification of the US conductors for ITER TF magnet system****Author:** Nicolai Martovetsky¹**Co-authors:** Wayne Reiersen²; Pierluigi Bruzzone³; Boris Stepanov³¹ *Lawrence Livermore National Laboratory, on assignment to Oak Ridge National Laboratory*² *Princeton Plasma Physics Laboratory, on assignment to Oak Ridge National Laboratory*³ *Swiss Plasma Center***Corresponding Authors:** reiersenwt@ornl.gov, pierluigi.bruzzone@psi.ch, nmk@ornl.gov, boris.stepanov@psi.ch

The US Domestic Agency is one of the six suppliers of the Toroidal Field (TF) conductor for ITER. In order to qualify conductors according to the ITER requirements, we prepared or provided to the Swiss Plasma Center (SPC) eight test articles, sixteen conductors total, that were tested in the SULTAN facility at SPC in the ITER relevant conditions. We also fully characterized the strands that were used in these SULTAN samples. In this paper, we report both test results and analysis of the conductors' performance against expected strand performance. The US TF conductors showed a little better than average current sharing temperature and a relatively low sensitivity to warm up-cooldown cycles in comparison with other suppliers. However, the trend in current sharing temperature versus cycles and warm ups did not saturate, which means continuing slow degradation of the conductor performance if number of warm up and cooldown cycles will be significantly higher than expected. The AC losses in the conductors are in line with losses in the other TF conductor suppliers.

This manuscript has been authored by UT-Battelle, LLC under Contract No. DE-AC05-00OR22725 and by Lawrence Livermore National Security, LLC, under Contract No. DE-AC52-07NA27344 with the U.S. Department of Energy. The United States Government retains and the publisher, by accepting the article for publication, acknowledges that the United States Government retains a non-exclusive, paid-up, irrevocable, world-wide license to publish or reproduce the published form of this manuscript, or allow others to do so, for United States Government purposes. The Department of Energy will provide public access to these results of federally sponsored research in accordance with the DOE Public Access Plan(<http://energy.gov/downloads/doe-public-access-plan>).

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 67 / 223**A construction design of helium recovery and purification system on HL-2M****Authors:** Xin Chen^{None}; Genliang Zhu^{None}; Hongbing Xu^{None}; Jinglong Chen^{None}; Youkun Fu^{None}

According to the needs of the development of China nuclear fusion research, HL-2M is being built in Southwestern Institute of Physics (SWIP) as a transformation and upgrade device of HL-2A, which is a plasma physics and controlled nuclear fusion tokamak experimental platform. The liquid helium cryogenic system (LHCS) with a 500W@4.5K power is under construction to provide a cryogenic and vacuum environment for HL-2M and its related equipments by total capacity of 7000L liquid helium.

So it's necessary to construct a helium recovery and purification system (HRPS), which contains a recovery part, a storage part and a purification part. The recovery part mainly includes 2 piston compressors with capacity 50m³/h and 100m³/h respectively, the storage part mainly includes 4800m³ higher purity helium tanks, 1200m³ dirty helium tanks and 50m³ gasbags, the purification system mainly includes a high pressure helium purification equipment, which consists of 2 purification storages, dryers and a helium purity analyzer. The maximum recovery rate of the system is designed to 150m³/h, and the helium purification equipments can improve the purity from 95% to 99.999% or even more.

Key words: HL-2M, LHCS, HRPS

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 23 / 224

Thermal Analysis and Test for the Mockup of ITER Radial X-Ray Camera

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Radial x-ray camera (RXC) is an important diagnostic device in the International Thermonuclear Experimental Reactor (ITER) tokamak. The function of x-ray camera is to provide a measurement of soft X-ray emission from plasma, which is necessary for the operation and control of plasma to support key physics researches. As a key device to measure the plasma in tokamak, it is required to meet the requirement of working temperature. In order to get the temperature distribution of the RXC in tokamak, the mockup of RXC is fabricated for test. In this paper numerical analysis is performed for the thermal loads on RXC mockup. The finite element model of RXC includes heat exchanger, detector, diagnostics shield module (DSM), cooling pipe, support and vacuum vessel. The heat source of RXC system comes from the high-temperature field of vacuum vessel, which causes temperature rise of RXC through heat conduction and radiation. To keep the RXC system in a reasonable working temperature, an efficient cooling system is designed. In the cooling system, helium is used as cooling medium to cool down RXC system through heat convection. The thermal analysis model combined of heat conduction, radiation and convection is performed for the conjugate heat transfer of RXC system. The analysis result shows a very similar temperature distribution of RXC with the test, which demonstrated the viability and accuracy of analysis method and test process and that present cooling system design meets the temperature requirement of RXC system. All the work done in this paper will provide an important reference for the RXC design.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 6 / 225

Investigation on the Effect of Tritium Production using Temperature Control for DEMO Blanket

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The tritium production is a key issue in the fuel recycle for DEMO blanket. It is affected by the temperature field inside the blanket interior due to the temperature requirement of the tritium release and the recovery. This paper discusses the tritium breeding ratio issues based on a PWR water-cooled blanket module. In particular, the variation trend of TBR is explored with the change of blanket interior. The tritium distribution is studied with the blanket temperature field. It is found that the pipe bore affects the local TBR sensitively, and the pipe bore with 9mm reaching the maximum local TBR for a blanket module. Tritium distribution calculations indicates there will be a large quantity of tritium generated in the area near the cooling pipes if the pipe bore is designed larger than 9mm. This will lead to the low tritium release efficiency for the blanket module due to the cooling effect of the pipes. Finally, the optimal range of the design parameters is obtained in view of TBR and tritium release performance.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 71 / 226

Numerical analysis of fracture behavior of first wall subjected to electromagnetic force during plasma disruption

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Plasma disruption would induce large eddy current in the first wall (FW) and other in-vessel components of the Tokamak system. With the huge confinement magnetic field in the Tokamak structure, huge electromagnetic force may generate in the in-vessel components. The study on the relationship between the plasma disruption and mechanical stress and strain in the FW and other in-vessel components can help to ensure that operation of the Tokamak system is confined to a region of operating space where threats to structural integrity are acceptable. Until now, many researches on the electromagnetic force in the FW without existing crack due to plasma disruption have been reported. However, if an initial crack exists on or near the surface of the FW, the flow of the eddy current would be disturbed, and the current density would increase at the crack tip. Therefore, more serious stress concentration would happen at the crack tip and may lead to crack propagation. To evaluate the fracture behavior of the FW with an initial crack under plasma disruption, a numerical model based on finite element method is developed to study the distribution of electric current and stress near the crack. A singularity of the current density distribution at the crack tip is observed. The stress intensity factor at the crack tip under different magnetic fields is applied to analyze the trend of crack propagation. Finally, the fracture behavior of cracks with different length and direction is also studied.

Eligible for student paper award?:

No

W.OA3: Neutronics and Multiphysics Analysis / 227**SuperMC Benchmark with SINBAD****Authors:** Lijuan Hao¹; Yongpo Lin¹; Qi Yang¹; Jing Song¹¹ *Institute of Nuclear Energy Safety Technology, CAS · FDS Team***Corresponding Authors:** qi.yang@fds.org.cn, yongpo.lin@fds.org.cn, lijuan.hao@fds.org.cn, jing.song@fds.org.cn

Development of nuclear engineering software should be based on a rigid software engineering process, in which, verification & validation process should be executed to assess the accuracy of the computed results against the corresponding benchmark values, and also to demonstrate the code performing adequately for intended applications. Super Monte Carlo Program for Nuclear and Radiation Simulation (SuperMC) is a general, intelligent, accurate and precise simulation software system for the nuclear design and safety evaluation of nuclear systems, and has been verified by more than 2000 benchmark models and experiments from the handbook of International Criticality Safety Benchmark Evaluation Project (ICSBEP), the Shielding Integral Benchmark Archive Database (SINBAD), and the International Reactor Physics handbook Evaluation Program (IRPhEP), etc., and also some other benchmark cases collected in FDS Team.

In this paper, the validation of SuperMC based on the benchmark archive of SINBAD was presented. Benchmark experiments performed on the D-T fusion neutron source facilities of OKTAVIAN, FNS, FNG, IPPE were selected to assess the shielding analysis capability of SuperMC for fusion engineering project. The materials of selected benchmark cases cover beryllium, graphite, oxygen, aluminum, silicon, iron, nickel, vanadium, tungsten, silicon carbide and stainless steel, etc., while the geometries of selected benchmark cases cover simple spheres, slabs and also complicated shield mockups for fusion engineering. All the selected benchmark problems were modeled and simulated with SuperMC. The calculation results, such as leakage spectrums, energy deposition, reactions rates and fission rates, etc., were compared with MCNP results and experimental data. Comparing to MCNP, very good agreement with deviations lower than 0.1% for integrated values of neutron spectra over energy was achieved, and the maximum deviations for induced photon integral flux are about 1% because photon transport models may differ slightly. Also, the neutron spectra results are in good agreement with the experiments.

SuperMC has been verified with D-T fusion neutron source benchmark cases from SINBAD, and excellent agreement has been obtained between SuperMC and MCNP, and also the calculation results are in good agreement with experimental data in most of the selected cases. The correctness and reliability of SuperMC modeling and simulation functions were fully validated.

Eligible for student paper award?:

No

M.OP3: Next Step Devices, DEMO, Power Plants / 228**Status of K-DEMO Design Concept Study****Authors:** Keeman Kim¹; Kihak Im¹; Sungjin Kwon²; Jongsung Park³; Hyoung Chan Kim²; Jun Ho Yeom²; George Neilson⁴; Charles Kessel⁵; Thomas Brown⁶; Peter Titus⁵; David Mikkelsen⁵¹ *National Fusion Research Institute, Republic of Korea*² *NFRI*³ *National Fusion Research Institute*⁴ *PPPL*⁵ *Princeton Plasma Physics Laboratory*⁶ *Princeton Plasma Physics Laboratory, Princeton University*

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The conceptual study on the Korean fusion demonstration reactor (K-DEMO) has been carried out since 2012. K-DEMO is featured by the medium size tokamak ($R = 6.8$ m, $a = 2.1$), a high magnetic field ($B_{T0} = 7.4$ T) with steady-state operation. The primary candidate of coolant medium is the pressurized water. One unique aspect of K-DEMO is a two-staged development plan to mitigate the gaps between the present level of technology and the required technology level for the full functions of DEMO. At first, K-DEMO targets not only to demonstrate a net electricity generation ($Q_{eng} > 1$) and a self-sustained tritium cycle, but also to function as a component test facility. Then, at its second stage, a major upgrade is expected to replace in-vessel components in order to demonstrate a net electric generation on the order of 500 MWe.

A preliminary operating scenario using a combination of various H&CDs (heating and current drives) covering neutral beam, electron cyclotron, lower hybrid, and fast wave H&CDs has been derived. The total H&CD power is estimated approximately 110 MW. The main components of K-DEMO have been conceptualized. The superconducting magnets (toroidal field (TF), poloidal field, and central solenoid magnets) were developed. 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 CICC (Cable-In Conduit Conductors) for each type of magnets were fabricated and tested. Divertor is adopting the monoblock-typed tungsten armors with the reference choice of a double-null operation. Solid ceramic pebble typed lithium orthosilicate (Li_4SiO_4) was primarily selected for the tritium breeder. Extensive mechanical and neutronic analyses have been carried out to support the developed design concepts and the results are presented.

1 K. Kim et al., "Design concept of K-DEMO for near-term implementation", Nuclear Fusion 55 (May 2015) 053027 (9pp).

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 76 / 229

A numerical model of RF ion source for the ITER-relevant NBI

Author: Xingquan Wu^{None}

Co-author: Jianglong Wei

With the development of magnetic confinement fusion, the new requirements and challenges are emerged for ITER NBI1. Briefly, It is required that the ion source of the neutral beam injection system should produce a uniform large volume high density plasma with the capability of long pulse steady state and long service life. Based on the EAST-NBI bucket ion source[2] where the main structure characteristics of large area high current ion source are introduced. In order to understand the radio frequency (RF) ion source this candidate for fusion NBI, here a numerical model of RF ion source is introduced, where the transport properties of electrons and ions are described based on the drift diffusion theory, The power coupling of RF power and plasma is analyzed, The influence of the external magnetic field on the plasma transport is also investigated.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 38 / 230

Experimental study on vacuum control method for Paschen tests of the superconducting magnet

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Abstract

In order to verify the integrity of the insulation of superconducting magnet, it is needed to perform the Paschen test on the insulation after its manufacture. The vacuum vessel for the Paschen test is required to keep for an adequately long time under the pressure values of different degrees. To achieve the low pressure of the vacuum vessel, isolating the pumping unit from the vessel is not applicable as degassing inside the vessel will eventually ruin the pressure. Therefore, one dynamic balance control system for low pressure control is designed. The feature of the system is that adjusting the opening of intake valve by Proportion Integration Differentiation (PID) control system automatically, while the vacuum pump is working constantly. The results show the dynamic balance control system can keep the pressure value of 1 ± 0.05 Pa, 10 ± 0.15 Pa, 100 ± 0.2 Pa and 1000 ± 0.5 Pa, respectively and the holding time of each vacuum degree is more than 2 hours, which satisfy the basic vacuum requirement for the Paschen test of the superconducting magnet.

Keywords: Superconducting magnet, Paschen test, Vacuum control, Dynamic balance, PID control system

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Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 42 / 231

Design and Analysis of CFETR CSMC Cooling Loop

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The Central Solenoid Model Coil (CSMC) of China Fusion Energy Test Reactor (CFETR) is currently in the design and manufacture process. CSMC assembly consists of the winding pack, an outer NbTi coil, a middle Nb₃Sn coil, an inner Nb₃Sn coil and a pre-load structure. The highest field of the model coil is 12T, while the highest change rate of magnetic field of the conductor is 1.5T/s. Due to the AC losses during charging, a huge heat load will be produced in the model coil. In order to make the coil work properly in normal condition, a well-designed and precisely-analyzed cooling loop plays an important role.

In this paper, the design of the cooling loops is based on the calculation results of the AC losses deposited on the model coil. The length of the cooling channels, together with the thermo-hydraulic parameters such as inlet pressure, temperature, mass flow rate are optimized. In addition, thermal

hydraulic analysis for the cooling loop located in the worst condition of the model coil was conducted to recognize the temperature and mass flow rate change over time. The hydraulic model, the material properties and the heat loads involved in the analysis are given, and the results of the analysis are presented.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 24 / 232

Multiple laser system for high resolution Thomson scattering diagnostics on the EAST tokamak

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The high temporal and spatial resolution of laser Thomson scattering (TS) diagnostics is an important research subject of fusion plasma diagnostics. Currently, the temporal resolution of TS based on single laser in EAST is limited to 20 ms, which is too low to resolve the evolution of pedestal structures directly. A critical part of this diagnostic is the high-frequency laser source. A multiple laser system for high resolution TS diagnostics has been designed and installed on the EAST tokamak in ASIPP (Institute of Plasma Physics, Chinese Academy of Sciences). To achieve the specified parameters, a multilaser solution including four 10-50 Hz 5 J Nd:YAG laser systems with the fundamental wavelength of 1064 nm, at a distance of ~40 m from the tokamak, is utilized. The design of the laser beam transport path is presented, using multi-beam combiner technology to improve the time resolution (up to 8 microseconds) and real-time monitoring of laser power to improve density measurement accuracy of the system. The requirements for this system are very stringent with approximately ~7mm spatial resolution at the edge region. After several weeks trial running on the superconducting EAST tokamak, the system was proved to be capable of measuring plasma electron temperature and density with high resolution. The setup of multiple laser system is described in detail in this paper, as well as the analysis of the measurement capability. Finally, the experimental results are presented. The completion of this project will provide the basic tools for other fast physical processes study, such as L-H transition.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 43 / 233

Test results about simple CDA+MIK quench detection method on EAST for ITER Superconducting CS Coils

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The test about simple CDA+MIK quench detection method implemented a total of about 100 charging runs on EAST in the past two years, aiming at verifying this new method whether apply it to

ITER cs coils or not. This project is supported by the ITER Organization (IO).

To obtain this target, the whole instrumentation hardware have been developed and installed on EAST and used to generate experiment data.

It is real challenge to achieve the plasma discharge due to CS module triad connection configuration, the limitation of peak coil current & voltage, very high accuracy high voltage measurement etc.

In verification experiments, the low loop voltage plasma discharge with the assistance of the electron cyclotron resonance frequency (ECRF) and low hybrid wave (LHW) heating for pre-ionization and to following burn-through was the first accomplished successfully plasma scenarios with around 3 second and 250kA pulse plasma current.

Moreover the numerical quench detection model have been designed and the artificial signals created through it should be verified with experimental signal.

The test have shown that CDA +MIK technology is applicable to the ITER cs coils as a backup method although the co-wound voltage tap sensor has obviously better noise rejection ratio.

But the sensitivity of this system should be improved greatly in order to meet experiment requirements for ITER operation in the future.

This paper introduces the test program, typical achieved operation, and the results of preliminary analysis.

Key words: CDA +MIK technology, EAST, verification experiments

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 77 / 234

0-D Physical Design for the Heating and Current Drive System of CFETR

Authors: Defeng Kong¹; Long Zeng¹; Xiaojv Liu¹; Hao Qu¹; Zixi Liu¹; Tao Zhang¹; Guoqiang Li¹; Xiang Gao¹

¹ ASIPP

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

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 25 / 235

Improving accuracy of interceptive current measurement for use in IFMIF/EVEDA accelerator

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ACCT/DCCTs are used as a non-interceptive means of beam current measurement in the IFMIF/EVEDA accelerator. Current measurement using ACCTs for long pulses with 100 ms or longer suffers such problems as drooping due to the fact that this measurement is based on the current induction through transformers. The electrical circuits connected to ACCT have been improved in order to reduce the drooping and to obtain a waveform as close as to the waveform of the original beam current.

In this presentation we propose a method for measuring the beam currents derived from the waveform of ACCT output signals. Since the ACCT and associated electrical circuits can be considered as a linear system, there must be a unique transfer function connecting the input and the output of the ACCT, etc. This transfer function and the backward transfer function can be obtained numerically from simple experiments. The conversion from the output waveform to the input waveform is "ideal" since they are free from restrictions of real circuits.

This method has several advantages: (1) no detail information about the ACCT and the electronics is necessary; (2) the transfer function is easy to obtain from simple experiments with a function generator and an oscilloscope other than the ACCT system; (3) effects of stray capacitance and inductance are inherently reflected in the transfer function; (4) the use of FFT speeds up the calculation for obtaining the transfer function. On the other hand, it does not allow a real-time beam monitoring since retrieving the accurate input waveform requires the whole waveform of the ACCT output.

For verification of this method, we conducted a set of simple measurements using a function generator, an ACCT and an amplifier to determine the transfer function numerically. Applying this transfer function to the ACCT output, the waveform of the original beam current was retrieved. In order to reduce the noise in obtaining the transfer function, a set of waveforms were carefully chosen so that the FFT window is not an integer multiple of the input pulse length.

For an FFT window of 3 seconds, five waveforms with pulse lengths between 0.7 – 1.3 seconds were used to determine the transfer function. The backward transfer function so obtained was applied to ACCT outputs with pulse lengths of 0.1 - 1.3 seconds. The retrieved waveforms show a very good agreement with the original square inputs without any drooping observed, showing the validity of this retrieval method.

In the presentation, we will show the theory and the procedure of retrieval as well as several results for waveforms with different kinds of shapes and lengths. We will also discuss potential problems and limitations of this method. Approaches to making this close to real-time current monitoring and implementation for use in the accelerator will be also discussed.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 28 / 236

Study on dynamic behavior of EAST upper divertor with vertical displacement events

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The upper divertor of EAST had been upgraded as ITER-like monoblock configuration after 2012 experimental campaign. Compared to previous graphite divertor, it can greatly improve handling capability which is capable of offering an opportunity to study ITER-related plasma physics and engineering and to study the performance of the advanced tungsten divertor at the same time. During plasma operation, the divertor will not only suffer complicated thermal load but also experience huge electromagnetic (EM) force. Considering that the EM forces induced by transient events like current quench (CQ) or vertical displacement events (VDEs) will deposit onto the divertor surface for several millisecond, it will bring a large damage to the divertor which maybe reduce the lifetime of the divertor or break the divertor directly. Eddy current induced by CQ and VDEs will greatly decrease since the monoblock structure is chosen as the upper divertor module. Therefore, halo current induced by VDEs which can reach up to about 50 percent of plasma current is the predominant source of the EM force. And it turns into one of the remarkable threats to capability of the tungsten divertor. Aiming at investigating the response and studying the dynamic behavior of the tungsten divertor components with VDEs, dynamical experiments on W and CuCrZr employed in EAST upper divertor had been done already. And based on the experiment results, constitutive equations involving five material constants were built by using Johnson-Cook model to describe the dynamical properties of these two materials. In this paper, three different halo current distributions on EAST divertor module were calculated respectively according to the halo configurations of vertical displacement events on ITER. Then the EM force and statics analysis were completed by importing the current results for each cases. Finally, dynamic behavior analysis on monoblock structure of the inner plates with 1 MA plasma current was performed and strength analysis and life prediction were done to evaluate the effects on divertors with VDEs based on dynamical properties gotten from the experiments and foregoing calculated EM force. This work was supported by National Magnetic Confinement Fusion Science Program of China (Contract No.2014GB101001).

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 10 / 237

The vacuum ultraviolet imaging system and its application on EAST

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The Chinese Fusion Engineering Test Reactor (CFETR) is the next device scheduled in the roadmap to realize fusion energy in China¹. It aims to bridge the gaps between ITER and DEMO. Steady-state operation is one of the key issues of CFETR. The EAST tokamak will provide a long-pulse, high power test bench for advanced scenarios under actively cooled metal wall condition, which will play an important role on supporting the steady-state operation of CFETR.

In CFETR, operation scenarios of H factor over ELMy H mode are around or higher than 1.0, as listed in Ref.1. For long-pulse ELMy H mode discharge, it is a big challenge for the divertor plate to hold the high-level transient heat flux due to the quasi-periodic ELM event. Therefore, ELM control is necessary to realize steady state operation. It is known that ELMs are strongly related with the dynamics of the so-called pedestal region, where steep pressure gradient exists. But the mechanism is still an open topic in fusion research. Experimental studies on the pedestal may be helpful on the understanding of the related physics and benefit the development of efficient method on ELM control.

A vacuum ultraviolet (VUV) imaging system is developing on EAST tokamak. It aims to measure the evolution of the spatial structures of the pedestal, by selectively measuring emission of 13.5 nm in wavelength, which mainly comes from C VI (one of the intrinsic impurities in EAST). It has been installed on EAST to view the plasma perpendicularly and has been operated in the 2016 experiment campaign. ELM dynamics can be studied by the combination of VUV imaging and the existing visible imaging system, which mainly monitors the bottom of the pedestal and SOL region on EAST. In this work, the hardware of the VUV imaging system and the first results from the VUV imaging data will be presented. In addition, the upgrade of the optics is scheduled for the next campaign, which can be operated to view the plasma tangentially. The proposals of the upgrade will be discussed as well.

Acknowledgement:

This work is supported by the Natural Science Foundation of China under Contract No. 11605244 and the National Magnetic Confinement Fusion Science Program of China under Contract No. 2014GB106000, 2014GB106001 and 2013GB106000.

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1 Y.X.Wan.et al 2016 26th IAEA Fusion Energy Conference, Kyoto, 17-22 Oct. 2016, Paper No. OV/3-4

Eligible for student paper award?:

No

T.OP3: Project Management and Systems Engineering / 238

Assessing Component Suitability and Optimising Plant Design – Alternative Approaches to TRLs

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Technology Readiness Levels are commonly taken to indicate the “suitability” of a particular component, process, etc. that is part of a larger system. Several versions exist; all assign a simple value (usually 1 to 9) defined in terms of the point in the development and testing cycle that the particular subject has completed. Thus TRL really reflects “maturity” rather than a holistic consideration of the interaction of the component with the system throughout its lifetime.

The TRL approach omits several aspects that are particularly important for fusion:

- probability and consequences of failure
- criticality of the component/process failure
- ability to manufacture and assemble parts correctly avoiding faults during operation
- variation of component suitability through its lifetime
- reliability, maintainability and through life cost.

Fusion combines many systems that use unique technologies and processes and testing these in a representative environment requires access to a fusion reactor. Inevitably, the first plants will have to be constructed from systems that have inherent risk attached to them and some of these will be critical to the operation of the plant (both in performance and safety). Although the TRL approach gives an indication of the level of technical risk associated with a technology, it does not provide a means to rank and manage that risk. In this paper we propose the adoption of Failure Mode Effects and Criticality Analysis (FMECA) to rank these risks and present some examples of applications to fusion.

FMECA, developed in the nuclear, automotive and aerospace industries, originated in the U.S. military. For each component/process all possible failure modes are ranked according to their frequency, detection probability and severity of consequence to the system.

For fusion this technique can be used to assess overall risk and criticality associated with a combination of different components and hence to make technology selection decisions based upon impact of failure rather than the level of maturity of a particular component (the TRL approach). The formulation of the assessment is also compatible with system engineering architectures used to identify component interactions, thus can be inherently part of the design process of a fusion reactor.

Early in the design process it can be used to:

- prioritise areas where R&D can deliver most benefit
- identify non-compliance with regulations
- define key inspection criteria
- define test and maintenance plans
- identify built-in test or failure indicators

Rigorous application to drive the programme will reduce the total time and cost to develop a DEMO as emphasis is placed on change early in the programme. A probabilistic approach to failure allows analysis to be extended throughout the lifetime of the reactor, reflecting the effects of component degradation, maintenance and replacement, thus rendering FMECA far more powerful and relevant to regulators and stakeholders than the TRL approach.

One issue of adopting this method for fusion will be the availability of knowledge and data on failure modes, particularly for the novel systems, and ITER will be of significant value in providing information to inform the analysis.

Eligible for student paper award?:

No

R.OP1: Diagnostics and Instrumentation II / 239

In situ and real time observation of tritium behavior in the metal by reversing associated particle spectra of DT neutron generator

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In the thermonuclear fusion reactor research, the tritium behavior in the metal, including adsorption, desorption, dissolution, diffusion, permeation, was critical to tritium economy and environmental safety, due to most material in the fusion reactor is metal, the tritium-metal interaction is a research focus in this field. The knowledge of the tritium behavior in the metal is a critical issue for the nuclear material research, it is necessary to perform tritium experiment due to there was no reliable and generous theoretical model for predicting tritium properties, for the purpose of evaluation of tritium amount and depth distribution contained in materials. While the strong demands for experiment data on the study of tritium in metals were suppressed by strict law and its costs. By utilizing the routine the associated alpha spectra in the DT neutron generator platform, a method which can in situ and real time observation of the tritium behavior in metal were developed. Based on the relationship between tritium profile in the target and the spectra of associated alpha, the reversion model for tritium depth profile was built, which the alpha spectra were used as model input. On the CIAE neutron generator, the alpha spectra in different tritium-titanium target phase were collected, through the built reversion model, each alpha spectra were converted to the tritium profile in the titanium, which presented the dynamic change of tritium distribution in the titanium. And the validity of these reversed tritium profile has been supported by the simulation result of the tritium-titanium interactions, which including the consideration of the isotope effects of tritium-deuteron. The presented was a cost-less methodology for real time acquiring the in situ tritium depth profile in the target material of D-T neutron generator.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 44 / 240

Study of electromagnetic effects induced by huge plasma current variations for EAST CS coils quench detection

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The quench detection for EAST superconducting CS coils is considered the most difficult quench detection work because of pulsed operation and the strong coupling with pulsed coils and huge plasma current. The coupling coefficient between superconductor and plasma is not fixed unlike the coupling with pulsed coils because the plasma current configuration is constantly changing. It lead to false quench triggers in case of big disruption due to ignore it in the original compensation system design which means a real challenge.

In order to discriminate inductive voltage induced by huge plasma current more thoroughly, the active plasma noise compensation system (APC) has been studied and developed on EAST tokamak. Due to the inductance between the plasma and the CS coils is time-varying with different plasma shape and density distribution, the main task of the APC is to get the dynamic compensation coefficient by calculating the time-varying inductance quickly and efficiently.

In the past few months, calculations and studies, taking different parameters (plasma shape, density distribution, fast plasma events etc.) into consideration, have improved voltage compensation greatly.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 35 / 241

ZrCo bed as Protium and Deuterium storage material

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It is a constant research interest in finding reliable solutions for the long term storage of hydrogen isotopes that integrates both the safety matters and its easy recovery. Thus, various methods have been investigated so far, namely gaseous storage in high pressure gas cylinders, liquid storage in cryogenic tanks or under solid state form. Considering as option the storage on solid substrate, for which a myriad of materials were investigated over time, this work focuses on the storage bed activation and absorption/desorption characteristics of a ZrCo bed alloy. Hydrogen isotopes (Protium and Deuterium) where selected as the working gases.

The activation was performed at 500 °C under vacuum conditions, hydrogenation at maximum 100 °C under a pressure of 1bar, and the dehydrogenation at temperatures up to 300 °C. Defined by qualities like easy activation, low equilibrium working pressure, high adsorption rate at low temperature and use of moderate temperature for hydrogen gas recovery, ZrCo represent one of the strongest candidate materials, beside uranium and titanium, for the design and construction of dedicated getter beds necessary to capture and transfer hydrogen isotopes

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 61 / 242

Multiphysics Modeling of the FW/Blanket of the U.S. Fusion Nuclear Science Facility (FNSF)

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The dual coolant lead-lithium (DCLL) blanket concept, which is utilized in the Fusion Nuclear Science Facility (FNSF) conceptual design, is based on a helium-cooled first wall and blanket structure with RAFS (Reduced Activation Ferritic Steel) and a self-cooled LiPb breeding zone. The objective of this work is to develop a multiphysics modeling process in order to optimize the design and achieve long lifetime, maintainability, and high reliability. 3D finite element multiphysics modeling of the DCLL first wall and blanket (midplane of one sector) has been performed using COMSOL 5.2. The multiphysics aspect of the design is demonstrated via coupling of Computational Fluid Dynamics (CFD), conjugate heat transfer and solid mechanics. Both normal and off-normal loading conditions have been analyzed. The results of velocity, pressure, and temperature distributions of helium flow, as well as the primary and thermal stress of the structure were obtained. This was followed by determination of the factors of safety along three critical paths based on the ITER Structural Design Criteria for In-vessel Components (ISDC-IC). We show here that the structural design meets the ITER-ISDC design rules under both normal and off-normal operating conditions, though the safety factors under off-normal condition with 8 MPa helium pressure are marginal. Thus simple design optimization was conducted based on a parametric study on first wall dimensions to improve the design.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 78 / 243

NBImag: a useful tool in the design of magnetic systems for the ITER Neutral Beam Injectors

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NBImag is a code suitable for the design and optimization of complex magnetic field configurations, such as that of a multi-aperture, multi-stage negative ion source and accelerator. The NBImag code has been developed for the design of the ITER Neutral Beam Injector (NBI), whose full-size prototype, MITICA, is presently under construction in Padova, Italy. The ITER injector shall produce a focused beam of neutral particles (H or D) having an energy of about 1 MeV and a total power of 16.5 MW, for 3600 s continuous operation.

The accelerator is constituted by a system of 7 conductive grids having different potential (from -1 MV to ground), each including 1280 apertures, with the purpose of forming a bundle of accelerated H- or D- beamlets with a total current up to 46 A or 40 A, respectively.

Since none of the available commercial (or freeware) codes was suitable for efficiently modelling such a complex magnetic field configuration with acceptable detail level and computation time, the code has been developed and used for optimizing the magnetic field configuration in the ion source and accelerator, so as to comply with the constraints existing in different regions:

- minimal magnetic field in the RF drivers for effective plasma start-up in the plasma source;
- reduction (filtering) of the fast electrons in the plasma source for efficient production and extraction of negative ions in proximity of the plasma-facing grid;
- optics quality (aiming and focusing) of 1280 negative ion beamlets ;
- disposal of co-extracted and stripped electrons and minimization of the heat loads on accelerator grids (by early deflecting co-extracted and stripped electrons).

A combination of a weak horizontal “long range” magnetic field produced by currents flowing in the plasma facing grid and in suitably arranged bus-bars, and of a strong vertical “local” magnetic field, produced by 5616 permanent magnets embedded in the accelerator grids, proved to be the most efficient configuration on the basis of an automated optimization procedure.

The NBImag code is based on an integral formulation and allows an efficient calculation of any static magnetic field configuration on the basis of the geometry of the magnetic sources, with linear material and permanent magnets. NBImag also includes magnetic force and inductance calculation, based on the same formulation. Thanks to the capability of efficiently describing a large number of permanent magnets with limited computational effort, NBImag has also been integrated with different automatic optimization procedures for the solution of inverse magnetic problems.

This paper describes the formulation of the code and of the optimization algorithms, the validation against analytical models and experimental measurements, and the application to the design of MITICA.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 29 / 244

Preliminary mechanism analysis of HyperVapotron experiment for high heat flux components

Authors: Delin Chu^{None}; Haiyan Jiang^{None}; Zhang Qianghua^{None}; Luoqin Mei^{None}; Baoguo Pan^{None}; Weihua Wang^{None}

The ITER Tokamak Cooling Water System (TCWS) is designed to provide cooling and baking for the Primary Heat Transfer Systems (PHTSs) and relevant systems including the first wall/blanket, vacuum vessel, divertor, etc. For its characters of promising to enhance the heat transfer performance and increase critical heat flux (CHF), HyperVapotron (HV) elements has been put forward as heat removal of high heat flux components in nuclear fusion research facilities.

In our study, a hypervapotron loop test facility was built to conduct some experiments of heat transfer. Phenomena of HV were observed using the techniques of planar laser induced fluorescent (PLIF), high speed photography, particle image velocimetry (PIV), etc. According to the design of the ITER cooling water system(CWS), the flow and condition parameters were utilized: (1) CuCrZr alloy material, (2) rectangle fin with height 8mm and width 3mm, (3) inlet subcooling temperature of 298K, (4) channel flow speed of around 6ms⁻¹, (5) maximum heat flux of around 1.5MWm⁻².

The relation of the heat transfer coefficient (HTC) between rectangle fin 3mm×8mm and rectangle fin 3mm×3mm will be presented under the identical tested conditions. Furthermore, the preliminary mechanism of the heat transfer will be explained by the coupling effect of the vortex flow with air bubble in the fin pitch, and the heat transfer efficiency is extraordinary dependent on the maintain time of vortex forming between the fins.

Eligible for student paper award?:

No

R.OP3: Tritium Extraction and Control / 245

Recent developments on the TRITON experiment

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The Permeation Against Vacuum (PAV) is the prevailing technology with regard to the tritium extraction from PbLi in the liquid metal-based breeder blankets. With the aim of achieving tritium self-sufficiency of the DEMO power plant, this technology allows the continuous operation mode in a relatively compact system by providing high extraction efficiency. A small scale prototype of PAV, TRITON, has been designed to experimentally test the capability of this technique. It will be installed in a PbLi loop in CIEMAT in order to demonstrate and validate its potential as tritium extraction system at DCLL-DEMO relevant conditions. TRITON is composed of hydrogen-permeable vanadium membranes and a stainless steel supporting structure, conforming rectangular channels for the flowing PbLi alternated with vacuum channels to extract the hydrogen.

In this work, the assembly strategy of TRITON is presented. Due to the involvement of materials with different properties the welding between all the parts is a critic point in the manufacturing process, requiring special attention. It is mandatory to reach a perfect sealing of the channels, therefore, separating the PbLi and vacuum sides. Hence, and prior to the experiments with PbLi, an exhaust analysis of leakages has been made in order to examine each join.

Once the structure is closed and installed in a dedicated experimental set-up, a series of experiments in gas phase have been carried out in order to check the extraction efficiency. This procedure is made at different temperatures thanks to a heating system located inside the prototype. Main results are presented for hydrogen and mixes with argon at various hydrogen partial pressures.

Eligible for student paper award?:

Yes

T.OP2: Fueling, Exhaust, and Vacuum Systems / 246**DESIGN OF CRYOGENIC TWIN SCREW HYDROGEN EXTRUDER SYSTEM****Authors:** Samiran Shanti Mukherjee¹; Ranjana Gangradey¹**Co-authors:** Devesh Tripathi ¹; Paresh Panchal ¹; Pratik Nayak ¹; Jyoti Shankar Mishra ¹; Jyoti Agarwal ¹; Krishanpal Patidar ¹¹ *Institute for Plasma Research, India***Corresponding Authors:** ranjana@ipr.res.in, samiran@ipr.res.in

Fuelling system is an important technological component of a fusion machine. With the advancement towards building up fusion reactors, plasma parameters like density and temperature are on rising scale. To have peaked density profile in such plasmas, fuelling by pellet injection has proved as efficient technology. India has its internal program for development of pellet injectors. The ingeniously developed SPINS-IND a single barrel pneumatic gun type pellet injector is successfully operating. The injector is able to freeze cylindrical pellets of size ranging from 1.8 mm to 4 mm. Pellet velocities achieved is a function of pellet size and propellant pressure. SPINS-IND has the achieved velocity range of 700-1000 m/s for 4-2 mm size pellets. Taking a step further development of twin screw cryogenic extruder system is undertaken at Institute for Plasma Research (IPR), India. The present extruder is a twin screw system comprising of in-line pre-cooler, liquefier and solid extrusion section. Solid hydrogen is produced and pushed forward with counter-rotating inter-meshing screws driven by servomotor capable of extruding sufficient solid hydrogen to get 3 mm (L) x 3 mm (D) size pellets at 10 Hz injection frequency. The concept of twin screw assembly, its support mechanism and stage wise cooling of hydrogen is discussed. Designed twin screw assembly is having screw root diameter of 28 mm with 10 mm screw pitch length, which can withstand a torque of 100 N-m. The screw cavity has rectangular cross section with inter-meshing angle of 54.53 degrees. In this paper the design methodology of screw elements will be discussed with its allied support to operate at cryogenic temperature. Design and analysis of screw gear, spline shafts and barrel with screw elements will also be presented.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 92 / 247**A ZCS AC/DC Converter with LCL****Author:** ZHU FENG^{None}**Co-author:** GE LI**Corresponding Authors:** lige@ustc.edu.cn, zhufengxy@126.com, hangqin@ipp.ac.cn

This paper studies an inductor-capacitor-inductor(LCL)voltage-source converter(VSC)which can be implemented by zero current switches(ZCS). The ZCS has potential applications in improving the efficiency of high-voltage high-current system, such as servo power supplies at over 10 MW for the compressed plasma suggested in[Li, G. High-gain high-field fusion plasma,Scientific Reports, 2015, 5]. This converter is composed of an ac/dc/ac insulated-gate bipolar transistor-based VSC converter and a passive LCL circuit designed for matching the conventional PWM converter. The converter parameters are tested for designing LCL circuit to have optimal response during the faults, realize ZCS control,and minimize converter losses.The simulation model is built based on the MATLAB platform.The simulation results show that the rectifier has good controllability and testing rig is being built.The simulation confirms the capability to control the load current for the compressed plasma .

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 11 / 248**Design of the optical emission spectroscopy diagnostic system and preliminary experimental results in RF negative ion source****Authors:** yan wang^{None}; zhi-min liu^{None}; jing-yan yan^{None}; li-zhen liang^{None}; jiang-long wei^{None}; chun-dong hu^{None}

The development of radio frequency (RF) negative ion sources for neutral beam systems requires knowledge of the plasma parameters. Optical emission spectroscopy (OES) is a non-invasive and in situ diagnostic tool, so optical emission spectroscopy diagnostic system are designed to be applied to the measurements of the RF negative ion source, and diagnostic principle and simplified analysis methods for plasma parameters are introduced. A preliminary results of a variety of plasma parameters are obtained based on the part of the local thermodynamic equilibrium (PLTE) state. When the discharge power is 25kW and the discharge operates pressure is 0.5Pa, the electron temperature is about 0.83eV and the positive hydrogen ion density is $2.7 \times 10^{18}/m^3$.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 30 / 249**3D numerical simulations of hypervapotron geometry on Thermalhydraulic Performance****Authors:** Ran Wei¹; K.C. Jiang¹; Qiang Li¹; Wanjing Wang²; Chunyi Xie²; X.L. Wang¹¹ ASIPP² Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP)**Corresponding Authors:** wjwang@ipp.ac.cn, xiecy@ipp.a.c.cn, weiran@ipp.ac.cn

In order to satisfy the EAST first wall and divertor upgrade plan, a hypervapotron (HV) cooling concept is chosen to be developed as a candidate for the design of PFCs. The HV structure relies on internal grooves or fins and boiling heat transfer to maximize the heat transfer capability. The fabrication technology of W/Cu divertor has been developed at ASIPP (Institute of Plasma Physics Chinese Academy of Sciences), and one W/CuCrZr/316L HV component will be fabricated for high heat flux tests. Before fabrication, the relevant analysis was carried out to optimize the structure of HV component element. In this paper, numerical simulaitons with a 3D model of 490 mm × 50 mm × 20 mm have been performed using the CFD (computational fluid dynamics) analysis by means of ANSYS FLUENT code. In the model, W tiles with thickness of 2mm were selected as armor tiles considering that 2-mm-thick W tiles are being used in EAST upper divertor. And two fin designs were compared for optimization, then the advantages of slots on the fins were also discussed. Besides, several width and shapes of the groove between the fin and the side wall were also compared. And for each design, the comparison between subcooled boiling and single phase convection has been carried out, as well.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 83 / 250

The offline simulation module of J-TEXT Real-Time Framework

Author: Yang Li¹**Co-authors:** Wei Zheng ²; Guozhen Zheng ¹; Ming Zhang ¹; Yuan Pan ¹; Ming Chen ¹¹ *Huazhong University of Science and Technology*² *Huazhong university of science and technology***Corresponding Authors:** zhengwei@hust.edu.cn, fx2003@hust.edu.cn

Tokamak as the most promising way to achieve fusion energy, the reliability of its plasma control system (PCS) is of great importance. J-TEXT Real-Time Framework (JRTF) is the next generation real-time framework for tokamak plasma control of J-TEXT. Introducing object-oriented programming (OOP) technology, JRTF is an advanced framework and can be a future candidate for developing PCS real-time framework of China Fusion Engineering Test Reactor (CFETR). For such huge device, efficient plasma operation and safety is essential. Thus, the complicated PCS needs to be tested to verify the reliability, robustness and availability before plasma operation. This paper proposed an offline simulation module based on JRTF to validate the PCS. OOP technology and flexible configuration using XML files contributes a lot to flexibility and the reusability of the offline simulation module. In the paper, the PCS of J-TEXT tokamak is briefly introduced and analyzed. Then, the conception design of creating offline simulation module's model is presented. In this section, we introduce a module to simulate the input data through history log and experiment data. The output is record by a simulated output module. Under such circumstance, the program is the same with the online program and only small changes in XML configure files is needed to import such offline simulation modules instead of real hardware I/O modules of reflective memory modules. In addition, a test in vertical field control system is conducted to proving the availability of offline simulation module of J-TEXT Real-Time Framework, an online experiment result has been compared with the offline simulation. Then, a test in horizon field control system is carried out in order to confirm the reusability of the offline simulation module. Also, a Simulink module converter has been developed. It can convert a Simulink designed module into a JRTF offline simulation module, which dynamically generates data based on Simulink designed models instead of using historical data.

Key Words: JRTF; OOP; offline simulation; J-TEXT; PCS**Eligible for student paper award?:**

Yes

M.POS: Poster Session M - Board: 53 / 251

The dynamic testing and analysis of copper and copper alloy in divertor working temperature range

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Abstract: Until now, the most advanced mature plasma facing unit (PFU) technology is the ITER W/Cu PFU, which is a monoblock structure composed by tungsten, copper and copper alloy (CuCrZr) as plasma facing material, interlayer and heat sink, respectively. Meanwhile copper and copper alloy play a role of structural materials in the divertor structure, on account of huge electromagnetic (EM) loads which are induced by eddy currents and halo currents in the magnetic field imposed on copper and copper alloy parts. The EM forces would give rise to material high strain rates because it can reach to a very large value. Moreover, sustained so high heat flux from high temperature plasma, temperature distribution of all divertor components is also complicated. In order to investigate the dynamic response of copper and copper alloy in divertors under EM loads, an experiment on CuCrZr used in EAST divertors was carried out employing a Split Hopkinson pressure bar (SHPB). As supplementary, quasi-static compression and tension tests on copper alloy and copper were performed to obtain basic stress-strain relationship using MTS hydro-servo system and DDL50 electronic testing machine separately. The true stress and true strain curves in relation to strain rates or temperatures are derived from above experiments, which would provide the necessary theoretical basis for evaluation on the dynamical response, fatigue life and damage evolution to divertor components under EM impact loads.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 12 / 252

Performance analysis on the VUV imaging system in EAST tokamak

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Co-authors: Tingfeng Ming ; Xiang Gao ; Yuming Wang ; Fan Zhou ; Feifei Long

Performance analysis on the VUV imaging system in EAST tokamak
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In the present fusion research, magnetically confined Tokamak device is one of the most promising candidates for future commercial fusion reactor. The Experimental Advanced Superconducting Tokamak (EAST), the first fully superconducting tokamak with D-shaped poloidal cross-section, can be operated under similar configurations with ITER. It aims at high-performance plasma for long-pulse operation scenarios under actively cooled metal wall condition. During the past few years, a lot of significant progress and advances in both physics and technology has been made on EAST tokamak. Additionally, studies on EAST will play an important role on both basic physics and key technologies for the Chinese Fusion Engineering Test Reactor (CFETR) as well.

A tangentially viewing vacuum ultraviolet (VUV) high-speed imaging system, based on an inverse Schwarzschild-type optic system is developing to measure the edge plasma emission (including the pedestal region) in EAST. The telescope system consists of two multilayer mirrors: a convex mirror and a concave mirror. The mirrors are made of layers of molybdenum and silicon, which can selectively reflect 13.5nm ($\Delta\lambda \sim 1$ nm) vuv light [2]. With this diagnostic, two dimensional (2D) structures of the edge magnetohydrodynamic (MHD) instabilities can be evaluated, which may be helpful on the physical understandings. In this work, the performance of this imaging system is discussed, including the image quality, estimation of spatial resolutions and noise levels, etc.

Acknowledgments

This work is supported by the Natural Science Foundation of China under Contract No. 11605244 and the National Magnetic Confinement Fusion Science Program of China under Contract No. 2014GB106000, 2014GB106001 and 2013GB106000.

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Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 7 / 253

The influence of heat transfer on MHD flow in the blanket at high Hartmann Number**Authors:** Congju Zuo^{None}; Jiajia Han^{None}; Weihua Wang¹; Haifei Deng^{None}¹ whwang@ipp.ac.cn**Corresponding Authors:** hjaza_f@163.com, congju_zuo@126.com

In the breeding blanket fusion reactor, the dynamic viscosity of liquid metal (LM) is influenced by heat transfer, leading to the change of velocity distribution. The effect of heat transfer on magnetohydrodynamic(MHD) flow in a rectangular duct at high Hartmann Number is investigated by a coupling method. In this method, the velocity field is calculated through a second-order projection method, coupling with the temperature distribution calculated by a finite volume method. The numerical result without temperature influence is validated by Hunt's and Shercliff's analytical solutions, and shows very good accuracy. On the basis of the numerical code, the velocity distribution of a Hunt's case with temperature influence is simulated. The simulation result indicates that the velocity field is different from the benchmark solution as the result of the influence of heat transfer.

Eligible for student paper award?:

Yes

M.POS: Poster Session M - Board: 89 / 254

Suppression of tungsten impurity by lithium injection in tungsten divertor on EAST**Author:** Wei Xu¹**Co-authors:** Ahmed Diallo²; Dennis Mansfield²; G.Z. Zuo³; Jiansheng Hu⁴; Kevin Tritz⁵; Rajesh Maingi²; Robert Lunsford²; T. Osborne⁶; Zhen Sun⁷¹ *Institute of Plasma Physics*² *Princeton University Plasma Physics Laboratory Princeton*³ *Institute of Plasma Physics, Chinese Academy of Sciences*⁴ *Institute of Plasma Physics, Chinese Academy of Sciences*⁵ *Johns Hopkins University*⁶ *General Atomics,*⁷ *ASIPP*

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EAST has upgraded the upper graphite divertor to ITER-like W/Cu monoblock structure¹ with active water cooling in order to facilitate the high power and long pulse plasmas^[2]. Without wall conditioning tungsten impurity accumulation has been usually observed in plasmas, which is a crucial impediment to achieving high power, long-pulse H-modes. Therefore, some wall conditioning technologies need be explored to suppress the tungsten impurity, such as lithium (Li) aerosol injection^[3] and Li coating^[4]. In 2016, plasma discharges are performed in tungsten (W) upper divertor, and some exciting results are obtained with Li aerosol injection.

The Li evaporation system in EAST has been upgraded with three new ovens located in the horizontal D, J, O port on EAST, separated toroidally by 120 deg.. The new ovens have three apertures for Li evaporation, for improving the Li coverage uniformity. In addition, there are two lithium powder dropper systems mounted in the J upper port: one located above the upper X-point, the other one located radially outboard between the X-point and outer midplane. The amount of injected lithium aerosol is controlled by a resonating piezoelectric disk.

The uniform Li coating with the new ovens effectively suppressed W impurity influx coming from W divertor to avoid impurity accumulation in the plasma core. Overall the Li coating provided an excellent wall conditioning for high performance plasmas on the W divertor, facilitating a 62s long H-mode. The real-time Li aerosol injection suppressed tungsten accumulation; plasma stored energy and confinement increased both in L- and H-mode. In addition the strength of the tungsten source decreased with the Li injection rate. Also, the inner target ion saturation current and electron temperature decreased at the inner target. Even after termination of Li aerosol injection, the core W intensity remained at a low level.

These results are encouraging as a possible mechanism to control tungsten impurities in future fusion devices.

Reference

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3. J.S. Hu et al., Phys.Rev. Letts. 114 (2015) 155001
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Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 111 / 255

Ultrafine Pt nanoparticles on superhydrophobic 3D graphene aerogel for hydrogen-water exchange reaction

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Hydrogen-water exchange reaction is one of the most useful methods to concentrate heavy hydrogen isotopes from an admixture containing hydrogen-rich gas and heavy water. Pt-based hydrophobic catalyst had attracted much research interest in the reaction between liquid water and hydrogen. The hydrophobic support prevented liquid water from covering the active sites of the Pt catalyst, hence the increasing stability of the catalyst. However, the weak adhesion between the metal catalyst and the support, the poor utilization percentage of catalyst and the lack of control on the catalyst morphology had seriously restricted the usage of the catalyst.

Herein, we reported that ultrafine Pt nanoparticles on superhydrophobic 3D graphene aerogel were prepared in hydrogen-water exchange reactions. First, 3D graphene aerogel was prepared as catalyst support via hydrothermal reaction. Second, monodispersed Pt nanoparticles were grown facilely on graphene aerogel through microwave assisted reaction serving as active catalyst. Finally, the aerogel was further modified by self-assembled monolayer to enhance the superhydrophobicity.

Our graphene aerogel supported catalyst showed good superhydrophobicity with contact angle larger

than 150 degrees, and strong adhesion between the Pt nanoparticles and graphene plane. The Pt catalyst was monodispersed and with particle size of 2 nm and was highly active for hydrogen-water exchange reactions. The catalyst was stable under experimental condition for more than one month. In addition, our catalyst was prepared using solution method during the whole process, which reduced significantly the cost and was beneficial to practical industrial use.

Eligible for student paper award?:

No

T.OA3: Blankets and Tritium Breeding: Liquid Breeders / 256

CONSOLIDATED DESIGN OF THE LOW TEMPERATURE EU-DCLL

Authors: David Rapisarda¹; Iván Fernández-Berceruelo¹; Iole Palermo¹; Fernando R. Urgorri¹; Luis Maqueda²; David Alonso²; Tomas Melichar³; Otakar Frýbort³; Ladislav Vála³; María González¹; Prachai Norajitra⁴; Heiko Neuberger⁴; Belit Garcinuño¹; Carlos Moreno¹; Marco Utili⁵

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A conceptual design of a DCLL outboard equatorial module was produced according to the specifications of the EU-DEMO based on 16 sectors, 1572 MW fusion power, with the main objective of bringing into maturity this breeder blanket concept. Specification and design guidelines for the DCLL Blanket System were developed, identifying the main requirements needed for the initial design and producing a preliminary CAD model of an equatorial module in a DEMO outboard segment based on neutronics, thermal-hydraulics and thermo-mechanical calculations. During the definition of this first conceptual design of the DCLL a new version of the EU-DEMO (with 18 sectors, 2037 MW fusion power) was released. Thus, the blanket design has been adapted to this new scenario by reviewing its operational conditions and producing important differences in the CAD model. Thus, some changes have been implemented with respect to the previous design, looking for simplicity. One of the most important ones is the new PbLi routing inside the modules, implemented to facilitate the draining of the individual modules. Related to this point, the previous annular geometry of the connection between the modules and the Back Supporting Structure has been simplified to reduce strong MHD problems. A comprehensive transient structural analysis revealed the occurrence of high stress concentration at the connections of the FW with the radial plates in case of a LOCA, therefore suggesting that an increase in the number of radial stiffening walls is necessary, and therefore the number of internal PbLi circuits.

Specific design elements have been consolidated, such as the thermal-hydraulic general scheme for the segments, the poloidal segmentation or the structural design. A MHD estimation of the convective heat transfer coefficient has been performed, and serves as input for the thermal-hydraulic and structural calculations.

Finally, an integration of the DCLL blanket within the PbLi loop is made, including the outcomes from tritium transport modeling in order to understand the overall behavior of the DCLL, as well as the impact on the Tritium Extraction System.

Eligible for student paper award?:

No

M.OP1: Plasma Operation and Control / 257**ELM pacing with lithium granules injection in W divertor on EAST****Author:** Zhen Sun¹**Co-authors:** R. Maingi²; Xiancai Meng³; Robert Lunsford²; Wei Xu³; Guizhong Zuo⁴; T. Osborne⁵; Ahmed Diallo²; Kevin Tritz⁶; Jiansheng Hu¹; Dennis Mansfield²¹ *Institute of Plasma Physics, Chinese Academy of Sciences*² *Princeton University Plasma Physics Laboratory Princeton*³ *Institute of Plasma Physics Chinese Academy of Sciences*⁴ *Institute of Plasma Physics CAS*⁵ *General Atomics,*⁶ *Johns Hopkins University***Corresponding Authors:** hujs@ipp.ac.cn, xuwei@ipp.ac.cn, sunzhen@ipp.ac.cn, mengxiancai@ipp.ac.cn, rlunsfor@pppl.gov, ktritz@pppl.gov, zuoguizh@ipp.ac.cn, adiallo@pppl.gov, dmansfield@pppl.gov

Pellet ELM pacing is a baseline ELM control strategy for ITER. While reliable and effective ELM pacing has been achieved by injection of deuterium pellets into carbon-walled tokamaks, the reduction of peak heat flux with high frequency ELM pacing in metal-walled tokamaks has been marginal. In comparison, the use of non-fuel pellets such as lithium (Li) is desirable for ITER due to its decoupling ELM pacing from fueling. For ELM control, Lithium Granule Injection (LGI) experiments have been carried out on EAST 1 and DIII-D[2]. The injection of sub-millimeter Li granules to trigger and pace ELMs has demonstrated heat flux mitigation on EAST and DIII-D, each with a carbon wall. In 2016, the LGI experiment was performed in tungsten (W) divertor on EAST, and some exciting results of LGI applications were obtained. ELM pacing efficiency was studied by injecting Li granules of nominal diameter 0.3–0.9 mm, with injected speed of 50–120 m s⁻¹. Robust ELM pacing with 100% efficiency of ELM triggering by Li granules injection was demonstrated in ITER-like wall plasmas during Li granule injection. ELM frequency was paced to ~130Hz from 300Hz, and high Z impurity accumulation was not observed. The experimental observations indicate that ELM triggering efficiency depends on many interwoven parameters, such as granule size, penetration depth, and heating scheme. Higher power discharges require larger granules for efficient triggering. A wide range of granule penetration depths was observed by two fast cameras. It was also observed Li granules injection shifted the density profile outward, which changed the characteristics of the edge fluctuations, i.e. more easily destabilizing the edge coherent mode. This work strengthens the basis for ELM pacing in future reactors.

This research is funded by the National Magnetic Confinement Fusion Science Program under Contract No. 2013GB114004 and the National Nature Science Foundation of China under Contracts No. 11625524 and No. 11321092.

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Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 93 / 258**A Novel Power Supply Design for Multistage Depressed Collector Gyrotrons****Authors:** Simon Keens¹; Marcel Frei¹; Andreas Epp¹

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Ampegon has been working with the Karlsruhe Institute of Technology (KIT) to develop a novel design of 10MW power supply for multistage depressed collector gyrotron tubes. These new tubes offer potential for greatly improved control leading to greater output efficiency when coupled with a capable power supply. Ampegon's new EPSM topology provides optimised control in two modes:

- HVDCPS (High voltage DC power supply) mode, continuous and pulsed operation, providing -90kV/120A accelerating voltage to the cathode with variable intermediate collector connectors providing up to -90kV cathode voltage. The system is capable to provide 10.8MW output power.
- PPS (Pulsed power system) pulsed operation, providing up to -130kV/120A accelerating voltage to the cathode.

The enhanced PSM topology is optimised to supply RF sources which must fulfil demanding phase- and amplitude-ripple requirements and handle high dynamic load impedances which is required to supply multistage collector gyrotrons. The basic topology is very similar to well-known PSM supplies. The major enhancement is achieved by including an additional DC/DC converter on the power module.

The power supply's variable voltage output function is necessary to improve degrees of control possible for the tube. The design offers the possibility to provide taps on the power supply to supply different voltages to the depressed collectors. Due to rotation of power modules on a standard PSM supply this was not previously possible. With the new EPSM design, modules supplying power are not rotated, and yet - with our novel design - the power supply still provides a level of stability beyond that required for gyrotron operation. With adjustable output voltage and flexible control, the power supply can modulate the cathode or collector voltages as required with square or arbitrary waveforms. The EPSM shows additional advantages compared to a standard PSM if dynamic loads (variable load currents but stable voltage) need to be powered.

The new topology, the project status and first test results are presented.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 8 / 259

Computational study of the elastic modulus of mixed pebble beds for WCSB

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Primary concept design of CFETR (Chinese Fusion Engineering Test Reactor) has finished. In the fusion reactor, tritium is bred by mixed pebble bed in CFETR's WCSB (Water Cooled Solid Blanket). In this study, the discrete element method (DEM) was used to study the mechanical behaviors and elastic moduli of mixed pebble beds. The effect of cyclic pressure p within the granular system in different sizes were considered. Besides that, we re-confirmed the nonlinear elastic stress-strain relation, the much lower elastic moduli of a granular system than that of solid materials and the faster growth of moduli than the $p^{1/3}$ law predicted by the effective medium theory (EMT). We also observed that the cyclic pressure (mechanical excitation) would stiffen the granular system, but this effect would be smaller as the cycle number increases. This indicates that the stability of system

could be stronger under this effect. Besides that, the subtle difference in grain sizes was observed to soften the system although it caused a little higher packing density. Furthermore, although the effective moduli of different granular materials E_s are diverse, they were found to nearly collapse to the different distribution when both E_s and p are non-dimensionalized by particle moduli E_p , and the mixed pebble bed cannot obey the EMT theory.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 45 / 260

Thermal-hydraulic analysis of high temperature superconducting magnets in CFETR

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The China Fusion Engineering Test Reactor (CFETR) is the next device in the roadmap for the realization of fusion energy in China, which aims to bridge the gaps between the fusion experimental reactor ITER and the demonstration reactor (DEMO). CFETR will be operated in two phases: Steady-state operation and self-sufficiency will be the two key issues for Phase I with a modest fusion power of up to 200 MW. Phase II aims for DEMO validation with a fusion power over 1 GW.

For saving the cost of construction and meeting both Phase I and Phase II targets with achievable technical solutions, a new design has been made by choosing a larger machine with $R = 6.6\text{m}$, $a = 1.8\text{m}$, $BT = 6\text{-}7\text{T}$. Over 1GW fusion power can be achieved technically and it is easy to transfer from Phase I to Phase II with the same machine. In order to obtain the maximum magnetic flux of 224 VS from the CS coils in Phase II, the high temperature superconductors of Bi2212 material are used for the CFETR reactor.[2]

In order to evaluate the feasibility of high temperature superconducting magnets used in CFETR, the 4C code is employed in this paper to analyze the thermal-hydraulic state of the coils.

The inlet and outlet pressure of helium cooling loops and operational temperature of the magnets is designed. The temperature margin of the superconducting magnets for the reference scenario of plasma discharge is estimated.

1 Yuanxi Wan, Jiangang Li et al , Overview of the present progress and activities on the Chinese Fusion Engineering Test Reactor, submitted to Nuclear Fusion.

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Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 13 / 261

Design and Analysis Progress of US ITER Integrated Diagnostic Equatorial Port 09

Authors: Yuhu Zhai¹; Russell Feder¹; Allan Besile²; Dave Johnson²; Wenping Wang²; Jingping Chen²; Andrei Khodak²; Mike Hause²; Han Zhang²; Jonathan Klabacha²; Julio Guirao³; Silvia Iglesias⁴; Victor Udintsev³

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ITER is the world's largest fusion device currently under construction in the South of France with over 50 diagnostic systems to be installed inside the port plugs (PPs), the interspace or the port cell region of various diagnostic ports. The Diagnostic First Wall (DFW) and Diagnostic Shielding Modules (DSM) are designed to protect front-end diagnostics from plasma neutron and radiation while providing apertures for diagnostic viewing access to the plasma. Three tenant diagnostic systems will be integrated into the equatorial port plug 09 (E09). The toroidal interferometer and polarimeter, or TIP system, is installed in the left drawer (DSM3) for measuring the plasma density so to control fuel inputs. The electron cyclotron emission (ECE) system is installed in the middle drawer (DSM2) to provide high spatial and temporal resolution measurements of the electron temperature evolution and electron thermal transport inferences. The visible/infrared wide angle viewing system will be installed in the right drawer (DSM1, right looking from plasma) to provide visible and IR viewing and temperature data of the first wall for its protection in support of the machine operation.

The PP engineering design and multi-physics analysis has been performed following ITER port integration requirements including weight limit (45 tons total), neutron shielding (100 uSv/hr total dose limit), cooling layout and structural integrity validation. Mass distribution for the TIP and ECE DSMs has been optimized to meet the weight limit by the new design of B4C shielding pockets. The lightened DSM maintains its front-end EM load distribution with better protection of on-board diagnostics; while still provides sufficient front-end stiffness for structural integrity. To moderate impact from VDE inertial loads due to the Vacuum Vessel (VV) movements during asymmetric plasma Vertical Displacement Events (VDEs), the rigid lock-in DSM to PP structure interface was implemented into the E09 port integration analysis models for design validation. The structural integrity of E09 PP assembly is largely driven by the electromagnetic loads induced on the metallic structural components during plasma disruptions. The in-port diagnostics and their mounting supports, on the other hand, are largely driven by the steady-state thermal loads from volumetric nuclear heating, and the dynamic response of components attached to the DSM-PP structure assembly under the VDE inertial loads. Progress on the E09 integrated design and analysis is reported. The tenant interface load transfer is also presented in details for in-port system attached to the DSMs as part of the design and analysis tasks for ITER PP engineering.

*This work is supported by US DOE Contract No. DE-AC02-09CH11466. All US activities are managed by the US ITER Project Office, hosted by Oak Ridge National Laboratory with partner labs Princeton Plasma Physics Laboratory and Savannah River National Laboratory. The project is being accomplished through a collaboration of DOE Laboratories, universities and industry.

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

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 112 / 262

Application of ZD REDOX Detection Technology for Measuring Hydrogen Isotopes in Tritium Extraction System

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It is limited for thermal conductivity detectors(TCDs) to measure hydrogen isotopes with gas chromatography in a large amount of helium gas environment. Then a new detection technology of ZD REDOX combined with chromatographic column was investigated in order to measure the low concentration of hydrogen isotopes in the atmosphere of tritium extraction system. The estimated detection limit for H₂/D₂ gas was 1 ppm in the mixed gases with 99.9% He, and the measuring precision of relative deviation was less than 5%.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 14 / 263

Upgrade of data acquisition and control system for microwave reflectometry on EAST

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Reflectometry system on EAST provides density profile and fluctuation measurement. The profile reflectometry worked under Q, V and W band, and the fluctuation reflectometry operated at fixed points in V band with 8 poloidal correlation channels. In present, an upgrade of reflectometry is being carried out on EAST. The U and E band profile reflectometry will be added into the current system. A new 4-channel poloidal correlation reflectometry working under W band will be developed. Owing the development of new reflectometry, the data acquisition and control system also need to be upgraded.

The upgraded profile reflectometry has 10 signal channels. Each signal channel requires a 14-bit 60 MSPS digitizer. So the total data rate is 1200 MB/s. A PXIe-based data acquisition system is designed to satisfy the requirement. Five dual-channel 14-bit 100MSPS digitizers (PXIe-5122) are used to digitize the signals. The generated data are transported via the backplane bus of chassis. The PXIe chassis (PXIe-1085) provides 16 slots and 4 GB/s dedicated bandwidth for each slot. A RAID0 disk array (HDD-8266) collects all the data from digitizers. The disk array consists of 24 SSD drivers and its maximum data rate is 3.6GB/s. A timing module receives the clock signal from central control system and distributes the clock signal to 5 digitizers for synchronization.

The new fluctuation reflectometry has 24 signal channels. The sample rate would be no more than 2 MSPS per channel, so the maximum total data rate is 96 MB/s. Three 8-channel 12-bit 60MSPS digitizers (PXIe-5015) are applied to digitize the 24 signals. The mechanical hard disk on the controller is replaced by SSD disk to attain higher writing data rate.

A dedicated 6-channel arbiter waveform generator (AWG) is developed to control the VCO module of profile reflectometry. The AWG outputs control signals with amplitude of 24V and data rate of 250 MSPS. The AWG can also receive clock signal from timing module to synchronize with central clock.

For remote control, control software for power supply is applied, when all the power supplies of reflectometry are connected via USB to the controller in PXI chassis. Reflectometry can be switched on/off remotely by hand, or automatically at a set time.

Now part of development work is still ongoing. According to the plan, all the system will be installed to EAST before the autumn/winter campaign of 2017.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 15 / 264

Design of a robust linear and rotary sensor compatible with hostile environmental conditions

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Large machines for fusion research and the fusion reactors have hostile environmental conditions characterised by high level of radiation, vacuum and temperature both in the vessel and in the port cell and the close area, commercial displacement sensors like linear and rotary encoders and their electronics are not suitable with this conditions. This problem has been approached and solved by developing an optical and optoelectronic concept for the reading section of rotary and linear encoders able to resist to the hard environmental conditions. In this concept the driving electronics of the system is placed far from the hostile environment and the connections to the sensor are realised by means of suitable optical fibers. Some of these devices was developed by us to instrument more complex systems, they have been also validated and tested in vacuum, temperature and radiation over long periods.

The paper describe a recent activity of design and simulation that has been performed in order to overcome the sole critical aspect of such methodology that is related to the very close reading distance between the static part and moving part of the encoder. The new design increases the reading distance of many times to relax the requisites of the associated mechanics thus making the system more reliable in case of large temperature gradients and levels. An upgraded concept of the optical and optoelectronics reading section has been conceived; being based on micro optics and micromechanics, before the implementation we have chosen to develop a custom 3D simulation tool both to validate the concept and to optimize the micro optical and micromechanical design. The tool is based on ray tracing algorithms developed using the 3D vector geometry. It was implemented using Mathematica and an optimization phase has been necessary to be able to perform simulations of hundreds of thousands of rays in useful time. Different choices in the design has been simulated in order to perform a trade-off analysis based on criteria of robustness, low cost, commercial availability of the components and depending on the environmental conditions. A brief description of the ray tracing algorithm is presented, followed by the numerical results obtained for the different choices of the configuration that have been evaluated in the trade-off analysis. The optimal design chosen for the implementation is discussed and presented in the paper and the expected characteristics are reported. An analysis of the possibility of increasing the resolution and the accuracy of the sensor by means of inter-mark interpolation is included. An outline of the driving electronics and its main characteristics is also presented showing that the new design optimizes also this part. The design is presented with the aim to show a suitable solution for displacement and rotary sensors, which can be adopted in diagnostics and/or in remote handling systems of large fusion machines or in other fields where similar hostile environmental conditions are present.

Eligible for student paper award?:

No

R.OP5: Experimental Devices II / 265

Design and Analysis of an Actively Cooled Window for a High Power Helicon Plasma Source

Authors: Arnold Lumsdaine¹; Steve Meitner¹; Richard Goulding¹; Juan Caneses¹; Dean McGinnis¹; Juergen Rapp¹

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The development of the next generation of magnetically confined plasma fusion facilities on the path to commercial fusion power will depend on an increased understanding of plasma material interactions (PMI). Plasma facing components in future facilities such as a Fusion Nuclear Science Facility (FNSF) or DEMO will experience high heat fluxes, high energy neutron fluence, and helium and hydrogen isotope permeation. This combination of environmental effects is unique to the fusion environment, and research and development in the field of PMI will require integrated facilities that can provide prototypical conditions. Particularly, facilities are required that will simulate conditions in the first wall and divertor regions. One such proposed facility is the Materials Plasma Exposure eXperiment (MPEX). MPEX will use a helicon antenna as a plasma source, which is a non-contact source that will produce a high-density plasma and minimize impurities. The plasma will be heated by electron Bernstein wave (up to 200 kW) and ion cyclotron heating (up to 400 kW) systems, which will produce heat fluxes of up to 10 MW/m² and ion fluxes of up to 1024/m²-s over a 75 cm² area at a target where the plasma will terminate. In order to provide long-pulse conditions, the plasma will be confined with superconducting magnets with on-axis fields from 1 to 2.5 T. All plasma facing components will be actively cooled. To examine the plasma interactions with neutron damaged materials, MPEX will have the capability to handle low activation irradiated samples.

The helicon source is based upon a design that has been successfully demonstrated in a prototype experiment (proto-MPEX). Power is coupled into the plasma through the antenna at a frequency of 13.56 MHz. The antenna is located in air, and the power is coupled through a ceramic cylinder (or "window") forming the vacuum boundary in this region. The antenna is located outside the vacuum due to the fact that high neutral pressures in the range 0.1 to 3 Pa are required in the helicon section in order to produce the required plasma densities, and at this pressure and power level antenna sputtering would otherwise be likely to occur that could contaminate material samples being tested. However, a drawback is that up to 20% of the power launched by the antenna is deposited on the inner surface of the window due to RF-plasma sheath interactions and the production of hot neutrals. The window thus must be adequately cooled so that thermal stresses do not become excessive. Experimental results from the uncooled window are correlated with finite element results in order to confirm the heat flux profile. The design of the actively cooled window is presented with computational fluid dynamics and finite element analyses to confirm the design will function with a 200 kW antenna within performance limits.

Eligible for student paper award?:

No

T.OP1: Power Supply Systems / 266

THE POWER SUPPLY SYSTEM OF SPIDER

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Each ITER Heating Neutral Beam injector (HNB) is designed to deliver up to 16.5 MW of additional heating power to the plasma, by accelerating Deuterium or Hydrogen negative ions down to -1MV with a beam current as high as 40 A in D- or 46 A in H-, for as long as 3600 s.

The performance required to this HNB, in terms of energy, power and beam-on time, goes far beyond the limits that have been reached in the HNB realized so far and presents many outstanding technical challenges. For this reason, it was decided to realize a Neutral Beam Test Facility, called PRIMA, located in Padova, Italy, at the Consorzio RFX premises, including two experiments: MITICA, the full-scale prototype of the ITER HNB injector and SPIDER, the full-size Radio Frequency (RF) negative-ions source, with the aim to contribute to make more efficient and reliable the future operation in ITER.

This paper gives an overview of the SPIDER Power Supply (PS) system, made in collaboration among ITER Organization, Fusion for Energy and Indian Domestic Agency. It is a special system, which includes the Ion Source Power Supply (ISEPS) to feed the circuits for the generation and the extraction of the negative ions, and the Acceleration Grid Power Supply (AGPS) to provide the high voltage (-96 kV) for their acceleration. Among design issues, key ones were the high insulation voltage level and the necessity to manage as normal operation the grid breakdown, a frequent and unpredictable event equivalent to a short-circuit of the AGPS load.

The ISEPS is hosted in a large Faraday cage, (HVD, 13 m × 11 m × 5 m) air insulated for -100 kV to ground, supplied by a main insulating transformer. A High Voltage Transmission Line, based on an original air insulated tri-axial design, provides the power and signal connections from the PS to the Ion Source.

The ISEPS is characterized by a variety of power supplies, including four RF generators (1 MHz, 200 kW) to produce the plasma, one high voltage generator (12 kV, 140 A) to produce the required electric field between plasma and extraction grids, one high current generator (15 V, 5 kA) to produce suitable magnetic field components that reduce the extraction of electrons, and other power supplies. The AGPS is rated for 96 kV / 71A; it is based on pulse step modulation technology and includes three oil insulated multi-secondary transformers and 150 switching modules.

The procurement, including stand-alone tests, of the HVD and ISEPS has been successfully completed and the AGPS installation is nearing completion. Once testing of the AGPS is finished, the next commissioning phase will start, aiming to integrate the operation of all PS units under the central control and interlock systems.

The paper, after recalling key requirements and main issues of the SPIDER PS system, will present the most significant aspects of the subsystems design, manufacturing, and testing and will report on the progress of the realization giving some highlights from the recent installation and commissioning activities.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 90 / 268

A Quasi-Periodic Linear Feeder for the Impurity Granular Injection on DIII-D

Author: Alexander Nagy¹

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Injection of solid non-fuel pellets has been actively used as a tool for pacing and mitigation of edge localized modes (ELMs). In DIII-D, effective ELM pacing has been demonstrated by high frequency injection of Li and C sub-millimeter spheres, using the Impurity Granule Injector (IGI) 1, which injects granules into the plasma at speeds up to 150 m/s, through impact with a rotating impeller. In the IGI, high frequency granule delivery was accomplished through a vibrational granule dropper, in which high time-average rates are obtained at the cost of lack of period control [2].

We present a new in-line granule feeder, capable of delivering granules of size 0.2-2.0 mm with no restriction of material properties, at quasi-periodic rates up to 150 Hz, for 0.7 mm diameter Li granules (600 Hz using 0.3 mm granules). The new dropper mechanism combines two piezo in-line units; one to feed, and one to circulate granules that are filtered out of the feeder path. A remotely adjustable filter eliminates granules that are stacked, oversized, or side-by-side to form a single moving granule injection line. The granules fall off the in-line feeder exit one at a time, hence achieving a quasi-periodic delivery at a rate proportional to the exit speed. At drop rates <60Hz, the granule delivery period has a variation of +/- 25%. At higher rates, the periodicity deteriorates. This behavior was studied using high-speed cameras and electrostatic measurements, and the variation appears caused by to gaps that develop in the last centimeter of the injection line, as granules exit off the moving track.

The linear feeder concept is robust against bridge instabilities and clogging issues, thanks to the simple diverter filter and constant recirculation of granules. Furthermore, the open-top design of the device allows easy access for refilling the device from separate reservoirs, and has easy access for directly monitoring operation and adjustment.

This paper describes the in-line feeder design details, along with several design iterations. The goal is a robust in-vacuum mechanism that can deliver flow ranging from a single particle to a line of particles at 150 per second, using different sizes and materials from the same apparatus.

1 Bortolon et al. Nucl.Fusion 2016

[2] Nagy Proceedings of SOFE 2015

This work was supported by the U.S. Department of Energy under DE-AC02-09C11466 and DE-FC02-04ER54698.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 57 / 269

Measurements and model calculations of activation reaction rate for (n,p) reaction on ^{54}Fe isotope

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In China Fusion Engineering Testing Reactor (CFETR) research, the blanket neutronics experiment is essential in validating the neutronics codes and tools used in blanket. The neutron activation method, supported by neutron transport calculations, is particularly useful in the estimation of the neutron intensity in the blanket, which based on the estimation of the activation reaction rate.

Due to Fe is a significant component of the structural material of blanket, including Water Cooled

Ceramic Blanket(WCCB), the accuracy of the reaction rate of $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ was investigated. We conducted activation experiment of Fe in the ^{252}Cf neutron field. The neutron source intensity of ^{252}Cf is $3 \times 10^8 \text{ n/s}$. After irradiated for 5 days, the activated Fe foil was measured by a high resolution gamma-ray spectrometer with a high-purity Germanium (HPGe) detector. The neutron flux calculations were carried out using Monte Carlo transport code and FENDL 3.0 Files. The neutron group cross section was calculated by NJOY.

The half-life of the product ^{54}Mn , 312.3d, is much longer than ^{56}Mn , so it should be taken into consideration in the radiation shielding calculation of the WCCB. The relative error between the calculated and experimental value is 9.72%. The main sources of the error are coming from the neutron source intensity, the calculation of HPGe efficiency, gamma full energy peak counts and etc.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 36 / 270

A new concept to achieve a higher fuel burn-up fraction in a DEMO reactor

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One of the key challenges for a fusion power plant is the need to increase the tritium burn-up fraction significantly from the values of only some 0.1% which result from extrapolation (at least above 5%). For a DEMO reference fusion power of 2 GW the fuelling rate necessary to replenish the burnt fuel is rather small ($\sim 2.7 \text{ Pa}\cdot\text{m}^3/\text{s}$) and the fuel burn-up fraction equal to the ratio of the burnt fuel to the particle throughput is small, indicating the need to maintain the lowest possible fuel throughput for reducing the required tritium inventory. The fuel throughput in DEMO is mainly determined by the necessity of He ash removal [1].

This paper proposes plasma bypassing from the divertor to the SOL region to ensure the He ash exhaust without required large D/T flow. As additional effect in case of a suitably chosen DEMO divertor configuration, He enrichment is expected because of the larger mean free path for neutral helium particles compared to the one of DT neutrals, which allow He to penetrate into the divertor region easier than DT atoms. These effects allow one to achieve the lowest possible fuel throughput for reducing the required tritium inventory. The divertor configuration with the dome and fuel bypassing from the plenum to the SOL is suggested as a promising divertor configuration which could facilitate He removal with moderate flow rates to be pumped and low tritium inventory.

[1] Yu. Igitkhanov, Ch. Day, P. Lang and B. Plöckl, paper to TOFE, 2016, Philadelphia.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 16 / 271

Research and design of microwave diagnostics on CFETR

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Microwave diagnostics including reflectometry and electron cyclotron emission are the candidates for tools to measure the basic parameters such as electron density and temperature profile on CFETR. They have high spatial and temporal resolution with the radial coverage of the entire plasma. Nevertheless, because of the transmission of signal relies on the waveguide which can survive in the neutron environment on CFETR, they have need for reduced access, front-end robustness. However, due to the high temperature of the target plasma especially in the core region based on the scenarios, realistic effect can change the position of the cutoff layers and downshift the ECE frequency. Based on the newly developed scenarios, the cutoff frequencies and the electron cyclotron frequencies are carefully calculated taking the realistic effect into consideration. The spatial coverage and resolution are evaluated under the developed scenarios of CFETR.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 9 / 272

NUMERICAL STUDY OF INTERACTION BETWEEN THERMAL STRESS OF THE FIRST WALL AND COOLANT DUCT BY LIQUID-SOLID COUPLED METHOD IN FUSION REACTOR BLANKET

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The He-cooled Lithium Lead breeder blanket could be developed with the utilization of relatively mature material technology, which is used by Reduced Activation Ferritic / Martensitic (RAF/M) steel as the structural material in China. It is necessary to analyse the first wall structure heated because of the thermal stress from two coolants in the blanket which would directly affect the blanket life and the safe operation coefficient, and indirectly carries on affection of the enhancement of thermal efficiency from electricity generation.

The helium flow in the First Wall and LiPb flow with a transverse magnetic field in vertical channels in the blanket are investigated. The specially numerical MHD code based on the CFD software has been developed for analysis of the LiPb flow. The helium flow with four kinds of design scheme have been calculated and simulated. The three-dimensional temperature distributions of the LiPb flow in heating duct have been given. The analysis of the flow field and temperature gradient in the boundary layer of the duct have been performed. The heat transfer boundary condition of helium flow duct was determined by means of liquid-solid coupled method. The analysis for the structural stresses of the LiPb flow channel have been performed. The effect of the ratio of thermal load on the heat transfer characteristics of the helium and LiPb flow have been calculated and performed.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 37 / 273

Cryopump development of the 5MW NBI system on HL-2M tokamak

Author: Xianfu Yang¹

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In order to carry out long pulse and high power NBI heating experiment on HL-2M tokamak, the high pumping speed pump is very necessary, which can rapidly remove the gas load from vacuum vessel, reduce the affect of the background gas in NBI injector on the tokamak plasma and ensure that the re-ionization loss of neutral beam in drift duct is less than 5%, so that a cryopump with large area interpolation plates based on three-stages adsorption structure has been developed. The design idea and size of three stages structure, the manufacturing technique and test experiment of simulated the actual operating mode for cryopump have been described in this paper. Experimental results show that the shielding plates can cool down to 80K when liquid nitrogen is added and cooling time is 1.5 hours, but the adsorption plates can only cool down to 120K by radiation from the shielding plates, the cooling time is 3.3 hours. The adsorption plates and the outlet of helium pipe can cool down to 5K and 6.3K respectively, when liquid helium is added, it takes only 10 minutes. This paper also shows three surface treatment processes to reduce the heat load of adsorption plates. In order to achieve good adsorption effect of hydrogen and deuterium, select the active charcoal whose specific surface area is 1923.92 square meters per gram and micro pore ratio is 94.7%, take twice bonding process, that the total amount of active charcoal can be reached 400 grams per square meter. And finally we obtain that the pumping speed of the cryopump is 2.3 million liters per second, the conductance probability of cryopump interface is 0.39, batter than that of pumps with chevron and louver structure shielding plates, liquid helium consumption is 11.3 grams per second, and several reasons cause of the increase in liquid helium consumption are analyzed.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 54 / 275

Study on the Welding Process of the Vacuum Vessel Mock-up for CFETR

Authors: ZHIHONG LIU^{None}; JIEFENG WU^{None}; JIANGUO MA^{None}; XIAOSONG FAN^{None}

Chinese Fusion Engineering Testing Reactor (CFETR) is a superconducting magnet Tokamak, The vacuum vessel (VV) will provide a high-vacuum environment for the plasma, improve radiation shielding and plasma stability and provide support for in-vessel components. The Research and Development (R&D) of the key technologies to the VV manufacture have been carried out a few years ago by Institute of Plasma Physics Chinese Academy of Science (ASIPP), including Narrow-Gap welding, cutting and non-destructive testing. The manufacture of the PS for first 1/32 sector VV mock-up has been completed in 2015, the 4 PS sectors have been welded into a whole in 2016. This paper will describe the study of the welding process of the 1/32 Vacuum Vessel Mock-up for CFETR.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 17 / 276

Optimization and Design of Divertor Langmuir Probe Diagnostic System on the EAST Tokamak

Author: Jichan Xu¹

Co-authors: Liang Wang²; Guosheng Xu²; Wei Feng; Huan Liu; Guozhong Deng; Jianbin Liu; Damao Yao; Guangnan Luo; Houyang Guo

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A flush-mounted Langmuir probe system has been built on the lower graphitic divertor targets on the EAST tokamak in 2016, which is transformed from the previous divertor Langmuir probes, aimed to reduce the erosion from high energy particle and strong heat flux on the probe surfaces exposed in the plasma, and explore new structure application. During the 2016 EAST campaign, the flush-mounted probe system has measured the plasma parameters by using single probe and triple probes respectively, to obtain electron density, electron temperature, particle and heat fluxes, and compared with the previous domed probe in the same plasma discharge condition. The results show that the plasma parameters measured by different measuring methods or different probe shapes are basically consistent, and demonstrate the flush-mounted probe system has been successfully used as a reliable diagnostic tool in the EAST divertor. Meanwhile, the design of the divertor Langmuir probe system is put forward and discussed, for the next generation of EAST lower divertor, which will be upgraded to full tungsten divertor with active water cooling, by optimizing the successful design of the Langmuir probe system on the ITER-like top tungsten divertor and the flush-mounted Langmuir probe system.

We would like to acknowledge the support and contributions from the rest of the EAST probe team, collaborators and individuals for design and fabrication of the divertor probe diagnostic system. This work was supported by National Magnetic Confinement Fusion Science Program of China under Contract Nos. 2013GB107003, 2015GB101000, 2013GB106000 and National Natural Science Foundation of China under Grant Nos. 11575236, 11422546, 11575235 as well as the Thousand Talent Plan of China.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 84 / 278

Comparison of radiative divertor behavior in Ar and Ne seeded plasmas in EAST

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In ITER and future fusion devices, a high radiation level for power exhaust will be mandatory to avoid thermal overload of divertor targets. Increasing divertor radiation by injecting impurities is a general and effective method to reduce scrape-off layer heat flux and to cool the divertor plasma to detachment. Impurities such as nitrogen (N₂), neon (Ne) and argon (Ar) have been widely used in radiative divertor experiments on several tokamaks. Last two years, Ar and Ne impurities were seeded respectively as the radiator from EAST upper divertor which upgraded into ITER-like full tungsten PFCs in 2014 to investigate their effects to plasma behavior, especially in the divertor region.

According to the cooling factor of Ar and Ne, which is closely associated with electron temperature, mixture of Ar/D₂ was firstly seeded from the upper divertor region as a radiator. To compare with Ar impurity, then experiments under similar plasma parameters' condition and the same gas puff position with Ne seeding, including pure Ne and Ne/D₂ mixture, were carried in 2016 campaign. In this work, both Ar and Ne impurity showed the high efficiency in reducing particle flux and heat load on divertor targets. After impurity seeding, saturation ion current, I_s , electron temperature, T_e , and heat flux on divertor target, q_t , decreased rapidly. In this case, the inner divertor first entered the detached state and the outer divertor followed the inner one soon. However, these two impurities showed clearly different radiation behavior. Compared with Ar impurity, the rise of radiation in Ne seeded plasma more located in the divertor region. It was more difficult for Ne to enter the plasma core region than Ar because the former belongs to a kind of low-Z impurity and has a lower cooling factor in the core region. After the gas puffing was terminated, it took 1~2s for the rise of radiation caused by Ar impurity to gradually drop down to the initial state, while the radiation after Ne seeding remained in a rising state until plasma burned out. The reason may be that Ar impurity quickly ionized but Ne impurity stayed in fluctuated ionization-recombination state due to the cooled plasma near the divertor target region where the low electron temperature as low as below 8 eV. With regard to impurities, there were notable increases and decreases of Li, C and tungsten impurity after the Ar impurity was seeded. These impurities, observed by the divertor impurity spectroscopy, dominated over all other kinds of impurity. However, these impurities presented a relatively low level in Ne seeded plasma. Therefore, it is indicated that using Ne as radiator preferentially in controlling PWI issue in radiative divertor experiments.

In addition, through Supersonic Molecular Beam Injection (SMBI) and divertor piezo valve collaborative control, we obtained the active feedback controlled radiative divertor operation last December. In this case, the rate of radiation loss, frac , could reach 40%, which is of great significance for the goal of long pulse high performance operations in EAST.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 1 / 280

Design and optimization of the CFETR breeding blanket with S-type cooling pipes in BU

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Three concepts of tritium breeding blanket have been proposed for Chinese Fusion Engineering Test Reactor (CFETR). One of the concepts is helium-cooled ceramic breeder (HCCB) blanket. The HCCB blanket have the S-type cooling pipes in breeding unit (BU), and the BU consists of lithium ceramic pebble beds and beryllium pebble beds. The breeding material and the multiplier material separated by the cooling plates. Theoretical calculations has been done to obtain temperature distribution and pressure drop for the preliminary design of the breeding blanket. Then ANSYS CFX is employed to verify the thermal performance of the first wall (FW) in radial-toroidal and poloidal-toroidal directions. The temperature distribution in FW is obtained and the optimization of cooling channels is proposed according to the simulation results. Finally, according to the temperature distribution, structural stress analysis has been done to verify the feasibility of the design.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 46 / 281**Structural Stress Analysis of the CFETR CS Model Coil****Authors:** Aihua Xu¹; Yu WU²¹ *institute of plasma physics Chinese academy of sciences*² *ASIPP***Corresponding Authors:** xuah@ipp.ac.cn, wuyu@ipp.ac.cn

CFETR (China Fusion Engineering Test Reactor) CS (Central Solenoid) model coil made with CICC (Cable in Conduit Conductor) superconductor had been developed in Institute of Plasma Physics, Chinese Academy of Sciences. The highest field of CS model coil is 12T, and the largest magnetic field change rate is 1.5T/S. CS model coil mainly consists of two Nb3Sn inner coils and three outer NbTi coils, buffer zone, feeders and joints, preload supports and so on. The inner diameter of the coil is 1500 mm, and the outer diameter is 3520 mm. Preliminary stress analyses were performed using coupled solver for simultaneous structural, thermal, and electromagnetic analysis. A global finite element model was created based on the initial design geometry data, and it was used to calculate the stresses and deformations of components. Numerical simulations were performed for room temperature condition, cool down to 4.5 K, and the operating current with 47 kA. Computational analysis led to the structural design of the coil, while the optimization was done during design process to verify structural integrity.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 79 / 282**Transport analysis of EAST long-pulse H-mode discharge with Integrated Modeling****Author:** Muquan Mu^{None}**Co-authors:** Bo Lyu ; Xianzu Gong ; Qing Zang ; Guoqiang Li ; Yanmin Duan ; Qilong Ren ; Siye Ding ; Kai Li ; Xiang Zhu ; Hui Lian ; Haiqing Liu ; Hao Qu ; Hongfei Du ; Chengkang Pan ; Xiang Gao ; Jinping Qian ; Jiale Chen

In the 2016 EAST experimental campaign, a steady-state long-pulse H-mode discharge lasting longer than one minute has been obtained using only Radio Frequency heating and current drive. Integrated modeling of one long-pulse H-mode discharge has been performed with equilibrium code EFIT, and transport codes TGYRO and ONETWO under integrated modeling framework OMFIT. The plasma current is fully-noninductively driven with a combination of ~2MW LHW, ~0.3MW ECH and ~1.1MW ICRF. Time evolution of predicted electron and ion temperature profiles through integrated modeling agree closely with that from measurements. The plasma current ($I_p \sim 0.45$ MA) and electron density are kept constantly and the simulated plasma current density profile is compared with that constrained by far-infrared polarimeter/interferometer. Validation with the experiments on EAST will increase our confidence for ITER and CFETR design and simulations.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 85 / 284

Application of laser-induced breakdown spectroscopy (LIBS) for in situ characterization of lithium deposition layer on EAST tokamak

Author: Zhenhua Hu^{None}

Co-authors: Cong Li ; Ding Wu ; Dongye Zhao ; Fang Ding ; Guangnan Luo ; Hongbin Ding ; Hongmin Mao ; Jiansheng Hu ; Jing Wu ; Liying Sun ; Ping Liu ; Zhen Sun

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Lithium wall conditioning has been a routine method to reduce fuel recycling and impurity deposition, which significantly improves the plasma performance in EAST tokamak 1. In the 2016 EAST experimental campaign, with the help of intensive lithium wall conditioning, one-minute steady state long-pulse H-mode discharge was obtained. However, the time and amount of lithium used for the daily wall conditioning were from previous experience. There are no effective methods for in situ and real time characterizing of wall conditioning situation on the first wall, especially the thickness and the local growth rate of deposited lithium layer as well as the hydrogen isotopes retention in the lithium layer. Laser-induced breakdown spectroscopy (LIBS) is a promising candidate for in situ characterization of the first wall. Recently, an in situ and remote LIBS system has been established for the first wall condition monitoring in the EAST tokamak [2].

In this work, the growth rate of the lithium layer was in situ and real time monitored by LIBS during the lithium wall conditioning. The results showed that the growth rate of the lithium layer was fast at the beginning of lithium conditioning and the growth rate becomes slower with time. According to post LIBS analysis in the laboratory, about 100 nm deposition layer ablated by one laser shot at the same energy density. About 2 μm lithium layer was estimated deposited on the first wall by lithium wall conditioning by 200 minutes in EAST. The thickness of the coating layer showed consistency with the amount of lithium for wall conditioning. The thicknesses of lithium coating layers were measured after wall conditioning and after a whole day plasma discharge for comparison. The results showed that about 500 nm lithium deposited layer was removed by EAST plasma discharges per day. The hydrogen isotopes were measured as well. The H/(H+D) ratio in the deposited layer after lithium conditioning was lower than that after EAST discharge, which means the deuterium was saturated with the reducing of the deposited layer and D-D discharge. The investigation of lithium layer and the hydrogen isotopes by LIBS in EAST will help to optimize and predict the wall conditioning for EAST operation and demonstrate the potential using LIBS in ITER.

1 Wan B., J. Li, H. Guo, et al., Nucl Fusion, 2015. 55(10):104015.

[2] Hu Z, Li C, Xiao Q, et al., Plasma Sci. Technol., 2017. 19(2): 025502.

Eligible for student paper award?:

No

M.OA1: Experimental Devices I / 285

Technical issues toward the steady state operation at KSTAR

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Fusion reactor needs the steady state operation and sustaining the long pulse operation beyond transient period in term of physics and engineering parameters is essentially one of key requirements in present non DT operation tokamaks.

Recently, KSTAR reported the long pulse operation beyond 1 minute at the injected power of about 5 MW and the plasma current of 0.5 MA. The normalized beta is about 1.5 and the total injected energy to the plasma reaches to about 300 MJ. It is shown that the bootstrap current fraction is below 40% and the discharge is interrupted by the surface temperature rise at in-vessel coil current other than physics issues.

In this talk, technical issues for extending to 100s operation with injected power of 12 MW and the plasma current of 1 MA in KSTAR are discussed in heating, plasma facing components and diagnostic system conjecting from present data of 60s operation and preparation efforts are shown.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 31 / 286

Thermo-Hydraulic Performance Testing for Plasma Facing Components by 3D Metal Printing Technology

Author: Suk-Kwon Kim¹**Co-authors:** Dong Jun Kim¹; Seong Dae Park¹; Hyung Gon Jin¹; Eo Hwak Lee¹; Jae-Sung Yoon¹; Dong Won Lee¹¹ Korea Atomic Energy Research Institute**Corresponding Authors:** skkim93@kaeri.re.kr, ehl@kaeri.re.kr, jhg@kaeri.re.kr, sdpark@kaeri.re.kr, jsyoon2@kaeri.re.kr, dwlee@kaeri.re.kr, kimdj@kaeri.re.kr

3D metal printing technology was selected for the development of fusion divertor research, and the optimization of thermo-hydraulic performance with a water cooling in a Korean heat load test facility by using electron beam (KoHLT-EB). The various cooling design for ITER and DEMO divertor have been fabricated for the enhancement of cooling performance, such as swirl tube and hypervapotron. The main target of this work is the overcoming of fabrication limitations in such cooling devices and the development of new cooling mechanism by using 3D metal printing. And 3D printed divertor mockup was designed and fabricated based on the optimization of 3D cooling structure. The high heat flux test facility KoHLT-EB was used to evaluate the enhancement of cooling capacities. KoHLT-EB was modified in water cooling system for the performance test and the experimental evaluation of the divertor mockups. High heat load for the divertor mockup was applied up to 10 - 20 MW/m². Also, Thermo-hydraulic and thermomechanical analysis with ANSYS-CFX were performed to determine the test conditions and performance of 3D printed mockups. Present research results will contribute the development of Korean fusion reactor and DEMO program.

Eligible for student paper award?:

No

R.OP5: Experimental Devices II / 288

Development of a utility negative ion test equipment with RF source at ASIPP

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The negative ion source is a key component of the neutral beam injector for the reactor-scale fusion devices, such as ITER, CFETR, and DEMO. Considered the lack of research on the negative ion source for NBI application in China, a utility negative ion test equipment with RF source is being developed at Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP). The original task of the test equipment is to train the design and operation of negative ion source at ASIPP. The further mission is to promote the research and development activities related to RF plasma discharge, magnetic field configuration, Cs dynamics, beam extraction, and beam optics.

The RF-driven negative ion source is being assembled, which is able to extract 120 H⁻ beamlets up to 60 kV for 10 s. The extracted current density is expected to 350A/m² with Cs-seeded. The test equipment has been completed, including the power supply system, the vacuum chamber, and the auxiliary system (e.g., cooling, pumping, gas). An alternative accelerator is also designed, where the electrode grids and insulators are immersed in the vacuum condition. Besides, a new electrical circuit topology is also applied for the alternative accelerator, where the plasma grid is at the ground potential, the last grid and beam dump are at the high potential.

Several diagnostic techniques are equipped to estimate and optimize the performance of negative ion source. In the source, the generation of negative ions is investigated with a set of electrostatic probes for plasma character, with optical emission spectroscopy for Cs and impurity density, and with cavity ring-down spectroscopy for negative ion density. The beam profile uniformity and divergence will be studied with beam emission spectroscopy, graphite tiles calorimeter, and beam dump with thermocouples array.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 113 / 289

Curent Status Concerning Tritium Removal Technology and its Implementation at Cernavoda NPP(ROMANIA)

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In CANDU fission reactors and also in fusion reactors, tritium should be recovered from large amounts of effluents for environmental and staff protection, for safety and for various applications. The combined cryogenic distillation(CD) with catalyzed isotopic exchange between deuterium and liquid tritiated water(LPCE) it's one of the most suitable technology for removal and its recovery. For LPCE process, the key issue and driver force consists of in a very efficient and stable contact element which has to work in direct contact with liquid and vapor water for long time with high separation performances. In order to check and to prove CD-LPCE technology for tritium removal from tritiated heavy water from Cernavoda CANDU Power Plant, an experimental pilot plant for

tritium removal (ExpTRF), has been built at ICSI Rm-Valcea and tested within comprehensive program.

Based on the authors' experiments and results, the present paper presents the current status and key aspects of activities concerning the operation of Tritium Removal Technology and its implementation at Cernavoda NPP in Romania. A comparison between present ExpTRF and the future Industrial Tritium Removal Facility (IndTRF) is shown and discussed.

The paper presents also a critical analysis on main contact elements used in LPCE module. The critical analysis it's focused on:

- selected types of hydrophobic catalysts and hydrophilic packing;
- methods and conditions for manufacture;
- key aspects in operation of TRF
- improvement of the performances of the proposed catalysts for industrial nuclear applications;
- extrapolation of research results at industrial scale;

As result, a new improved contact element, more compact, has been developed and it's still under testing at ICSI Rm-Valcea. This new improved contact element has been selected to equipped the LPCE column within Industrial Pilot Plant for Tritium Removal Facility at Cernavoda NPP.

This new improved contact element could be an option in the process of selection of catalytic mixed packing for Water Detritiation System(WDS) and Isotopic Separation System (ISS) from the ITER reactor.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 55 / 290

Reverse Processing of CFETR Vacuum Vessel Mock-up

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The 1/32 sector of CFETR vacuum vessel mock-up is welded together with four poloidal segments (PS), which are made by forming and welding. So each PS has certain machining allowance and contour error, lacks the processing datum in the process of machining welding groove. Considering each PS has complex surface profile, the method of reverse engineering is adopted to get the 3D model of each PS which has been made. On the basis of this 3D model, the model for CNC machining can be designed. 3D datum transformation is used for solving the problem of lacking the processing datum. An additional part with regular shape is used for converting the model for CNC machining into the machine tool coordinate system. Now, four poloidal segments have completed the welding groove processing. The machining deviation of welding groove is less than 0.5 mm, which has meet the requirements of assembly and welding.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 109 / 291

Structural Integrity Report of Neutron Flux Monitor at occluded EqP#07 (PBS 55.B4.D0)

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ITER is one of the most ambitious energy projects in the world today. The Neutron Flux Monitor (NFM) diagnostics module will be installed on ITER, it measures the total neutron emission, providing the evaluation of the fusion power, and will be positioned in the occluded Equatorial Port #07, more exactly mounted on the inner Bio-shield wall and within penetration from Tokamak Pit to the NB Cell.

This article describes the overall design of NFM, defines relevant failure modes, criteria of the structural integrity assessment, gives an overview of the structural design criteria used in the structural integrity assessment of the ITER NFM#07 system. In the article, NFM is analyzed by the finite element tool ANSYS and the simulation result is evaluated based on analytical design, then summarizes the structural assessments related to each failure mode and the further research about the stress distribution at the large stress intensity area is performed.

The results and outcomes of this assessment will be taken into account for the design and the future fabrication of the NFM#07 system and its components.

Eligible for student paper award?:

No

M.POS: Poster Session M - Board: 56 / 292

Application of Contour Fitting Method in CFETR VV Assembly

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China Fusion Engineering Test Reactor (CFETR) vacuum vessel (VV) is a forming and welding product, which is composed of formed shells and ribs, and welded to a toroidal double shell structure device. However, there are manufacturing errors in the manufacturing process, which will lead to accumulative deviation of the VV dimensions and affect the VV contour and the welds quality. To reduce and decentralize errors, the contours of welded parts are measured and fitted with least-squares method. According to the fitting result, new benchmarks are reset in form of controlling points for the next assembly. This paper is about the application of contour fitting method in manufacturing of the first 1/32 VV sector of the CFETR 1/8 VV mock-up.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 2 / 293

Classification of TBM components for construction code application

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Test Blanket Module (TBM) system in ITER facilities shall be designed, fabricated, and installed according to the construction code appropriate to the component class of TBM system. It is essential to properly define the component class of the system considering safety, quality, seismic and etc. Current construction codes have been well established and applied to nuclear power plants. However, due to the difference on the criterion about the classes, it is unclear to apply existing construction codes based on nuclear power plant condition in case of not only ITER conditions but fusion power generation conditions. In this paper, the criterion of each class for nuclear power plants and that for TBM system among ITER facilities are compared and the differences in each criterion are summarized. The component class of each component in TBM-set, which is located at the front of TBM system and has major function in TBM system, is established per each criterion and the applicable construction codes for each component are determined considering component class and operating conditions. Construction codes including ASME and RCC-MR (RCC-MRx) are used in this paper.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 80 / 294

Analysis and experimental study of impedance matching characteristic of RF ion source on neutral beam injector

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The neutral beam injector (NBI) is one of the plasma heating methods on fusion device, which has highest plasma heating efficiency and the clearest heat physical mechanism. The high power ion source is one of the key parts of NBI system. Compare to the traditional hot cathode ion source, the radio frequency (RF) ion source have many merits, such as higher lifetime because of no filaments, simpler mechanical structure, lower cost due to the cheaper power supply, and power supply on ground potential due to a transformer used. It is also the reference ion source for ITER. The impedance matching is the important unit for the RF ion source, which is used to match the parameters of the RF generator and ion source antenna. It can helps to transfer the maximum RF power to the RF antennal of ion source and gets stable plasma. Due the plasma impedance will be changed before and after the plasma generation, the impedance characteristic is not easy to be calculated and measured. So, it also need more experimental study. In this paper, the impedance matching unit was analyzed and designed according to the principle of RF ion source. The matching characteristic was studied during the experiment, and got the best impedance matching characteristic. It also verified the design of impedance matching unit. Based on the results of impedance matching study, high RF power of 50 kW was coupled into the plasma and got long pulse stable plasma discharge.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 32 / 295

Thermomechanical Assessment of the K-DEMO Divertor Target Applying CuCrZr and RAFM as Heat Sink Materials

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Divertor is one of the most challenging and important components in DEMO plants, since the enormous heat load from plasma applied onto the divertor target must cool down. In a conceptual study of the Korean fusion demonstration reactor (K-DEMO), a water-cooled divertor concept applying the tungsten monoblock type was of primary consideration. The target peak heat flux of 10 MW/m² was set in steady state operation. To faithfully cool down the heat load, the selection of materials that the divertor is composed of is important as well as the decision of design parameters. Especially, the choice and design of the heat sink material in the divertor target are quite significant because the heat sink directly interfaced with the coolant. Reduced activation ferritic martensitic (RAFM) steel and CuCrZr have been considered the most promising candidates as the heat sink material. The preliminary designs of the high heat flux (HHF) units operating within materials' own allowable temperature were derived by accomplishing thermohydraulic analyses for RAFM and CuCrZr. Based on the designs of HHF units with a support structure, thermomechanical analyses were carried out. In mechanical analyses, the mechanical loads including the body force, the pressure caused by the coolant, the electromagnetic force were considered as well as the thermal load imported from computational fluid dynamics calculation. In this study, the structural stability of the divertor target applying RAFM and CuCrZr heat sink was accessed by performing the elasto-plastic analysis

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 94 / 296

An Active Gate Control for Press-Pack IGBTs in Series applied for high-voltage switch

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Due to safety and reliability concerns, fast switch is required for the J-TEXT ECRH system to cut off the 100 kV power supply within 20 μ s when fault occurs. For its high switching speed, excellent controllability and small size, IGBT is an ideal choice as the basic component of the switch to satisfy the fast response requirement. Obviously, connecting multiple IGBTs in series is an essential approach in high-voltage applications. However, due to the on-state characteristic deviations of each IGBT and stray parameters of the main circuit, static and dynamic overvoltage problems may occur. It will damage the IGBTs, even gives rise to device failure. So voltage balance between the IGBTs connected in series is the main task in the design of the switch.

This paper presents an active gate control method for IGBTs in series. It limits the dv/dt on IGBT to inhibit overvoltage by multi-changing gate resistor. In the meantime, the trigger time of each IGBT would be compensated for voltage balance purpose. In addition, considering the significant advantage of Press-Pack technology in series application, Press-Pack IGBT (PPI) has been applied to enhance the reliability of the switch. The test prototype is a 5000 V/150 A switch with 4 PPIs in series. The simulation and experimental results show that the method can effectively improve the dynamic-voltage balance. All in all, the work presented in this paper has a significant reference value for practical engineering applications.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 115 / 297

A Maximum Current Control Strategy for Three-phase PWM Rectifier for the ITER In-Vessel Vertical Stability Coil Power Supply

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The required peak current of ITER in-vessel vertical stability (VS) coil power supply is up to 80 kA, so VS coil power supply needs a PWM rectifier to achieve high power factor operation under the highly transient power demand. A new maximum current control method for three-phase PWM rectifier based on its mathematical model in d-q coordinate has been discussed. The control method samples DC-voltage of power supply and changes the set-value of current-loop controller instantaneously at different voltage values, it meets the fast-charge demand of power supply and achieves a unity or high power factor operation. The feasibility of the control method has been verified by simulation and experiment.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 71 / 298

Experimental Study on Multilayer Liquid Metal Film Flow Characteristics under Horizontal Magnetic field

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The liquid lithium has been considered as a suitable selection for the plasma facing component (PFC) materials in fusion reactors because of its plenty of advantages for removing the heat fluxes, incident tritium and impurity effluxes, and enabling a lithium wall fusion regime. However, under a complex magnetic field, the flowing liquid metal will exhibit complicated flow characteristics induced by the action of extra Lorentz force, which named magnetohydrodynamic (MHD) effect. Some preliminary experimental and numerical studies have proven that the flow resistance increased dramatically, the surface wave of film flow changed greatly, and liquid film cannot cover the whole solid surface with the existence of magnetic field. Because of the above deficits, it is hardly to form a uniform, stable lithium film in the Tokamak environment. In this paper, we make an attempt to make up the disadvantages on liquid metal film under magnetic field. A newly built liquid film generator with eight outlets distributed in different heights to form eight layers short liquid films, which connect one

after another to form a long liquid film, is used to test the feasibility of our idea on the production of ideal liquid film. The Galinstan, at liquid state in room temperature and low toxicity, is chosen to substitute the lithium in our experiments. Experimental results show that this kind of film generator can enhance the spreading performance of liquid metal on solid surface and reduce the flow resistance induced by a magnetic field. A preliminary analysis is also carried and an evaluation of using this kind of liquid film generator as PFC in real fusion device has been conducted to lead some further studies.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 82 / 299

Research on Synchronous Data Network of J-TEXT Plasma Control System

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PCS is one of the key systems in Tokamak. PCS contains many subsystems, which can be used to control the different plasma parameters quickly and effectively. There is a need for real-time communication between the subsystems of PCS, and it has high requirements on the delay and stability of data transmission.

In this paper, the overall design of PCS Synchronous Data Network is given. The network uses two kinds of technologies, hard Synchronous Data Network based on the reflective memory (RFM) network and soft Synchronous Data Network based on the 10G Ethernet network. RFM network data synchronization delay is deterministic and predictable, so it is more stable and reliable, but the cost is high and not as flexible as Ethernet. Ethernet data synchronization delay is not as deterministic as RFM network, but with 10G Ethernet the delay is small enough to be regarded as deterministic thus can be used for real-time control. It is also cheaper, more flexible and has higher throughput.

In this paper, transmission delay between nodes in different network with different kinds of load are tested under the new plasma control system based on JRTF. The test results show that the two kinds of the Synchronous Data Network which have been integrated into the new generation PCS system, can both satisfy the real-time control tasks, and enhance the stability of the power supply systems. But each with different characteristics, and J-TEXT PCS has chosen RFM network as its SDN network.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 49 / 300

The Influences of irradiation defects on mechanical properties for ceramic breeder material Li₂TiO₃

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Tritium breeder materials are significant for blanket design of fusion reactor. However, during blanket operation, the ceramic breeder materials will be subjected to neutron irradiation which could be detrimental to mechanical properties. Because of its good chemical stability and available tritium release behavior, Li_2TiO_3 is becoming one of candidate ceramic breeder materials.

In this study, Li_2TiO_3 samples are irradiated by 120keV deuterium ions. For sample characterization, the phase composition is investigated by using X-ray diffraction (XRD) before and after irradiation. After deuterium irradiation, the Electron spin resonance (ESR) experiments are employed to investigate the irradiation defects. Micro-hardness measurement is applied to study the changes of mechanical properties. XRD results indicate that Li_2TiO_3 crystals are damaged by deuterium irradiation, but no new phases are produced. According to ESR experiment, the defect type after deuterium irradiation is E-center which are vacancies trapping one electron. From Vickers hardness measurement, size effect of micro-hardness is observed. The Meyer coefficient obtained in the experiment is 1.65 which is less than 2. The Vickers hardness increases as applied loads decrease which are consistent to Meyer theory. And the Vickers hardness of the Li_2TiO_3 decreases as irradiation doses increase. The details of this condition are under investigation.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 72 / 301

Experimental Study on the Liquid Lithium Film Flow characteristics under Spanwise direction Magnetic field

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Using liquid lithium film as the plasma facing component (PFC) is the prospective scheme in the future magnetic confinement fusion to withstand the plenty heat flux and improve the plasma performance. Under the magnetic field along the film spanwise direction, the lithium film flow will exhibit a complicated flow characteristics induced by the action of extra Lorentz force, which named magnetohydrodynamic (MHD) effect. Because of the lithium physical properties (e.g high melting point, high chemical activity and so on) and the space limitation of the magnet, some preliminary experimental studies are carried out by using the room temperature liquid metal Galinstan, which proved that the spanwise magnetic field could thicken the film, suppress the film flow turbulence, detach the flow away from the side wall, change the film surface wall, and so on. However, these experimental results can not apply to lithium entirely because of the great differences in physical properties (i.e the density of the lithium is small than one tenth of the Galinstan). In the present paper, the experimental facility of liquid lithium film flow under uniform spanwise magnetic field has been established at UCAS (University of Chinese Academy of Science), in order to study the spanwise magnetic influence on the lithium film behaviors. The test section, which is made by stainless steel, with the upside visible, enables to produce the lithium film flow with a width of 60mm and inclined angle of 60. The flow rate, which is driven by the high pressure argon, could be adjusted from 0 to 10 L/min. The strength of the external magnetic field, which is generated by an electromagnet, is varied from 0 to 2T, with maximal unevenness lower than 5%. Experimental results show that the lithium film flow in the stainless-steel test section is significantly changed by the spanwise direction magnetic field, the surface waves are suppressed and became more stable. Finally, some quantitative analyses are also carried in present paper.

Eligible for student paper award?:

No

M.PLN: Plenary M / 302

CFETR- New Design and R&D Activities

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Roadmap of MFE research in China has been discussed by MF research community after CN joint ITER. The most consensus conclusion is showed in figure 1. The key step after ITER for FE development in China is CFETR. It will be the next key device for CN MFE program and will aims to bridge the gaps between ITER and the demonstration reactor DEMO.

Mission and objectives of CFETR: 1) a good complementarities with ITER; 2) demonstration of full cycle of fusion energy; 3) demonstration of full cycle of T with TBR over 1.0; 4) long pulse or steady-state operation with duty cycle 0.3 ~ 0.5; 5) develop the new advanced technologies such as diagnostics/control for burning plasma, CW H&CD, materials, RH etc for DEMO.

Status of CFETR project: 1) the first concept design has been completed; 2) a new design is under developing; 3) some important R&D supported by different channels has made important progress; 4) further budget support for engineering design and R&D activities will be provided by government soon possibly.

- Both first design and the new design will aim to operate in two phases. Steady-state operation and tritium self-sustainment will be two key issues for the first phase with a modest fusion power up to 200 MW. The second phase aims validation for DEMO with a FP over 1 GW.
- Advanced H-mode physics, higher TF magnetic fields up to 7T, CS coil up to 14 T, larger size (R, a/b), less number of TF magnets for higher accessibility, high frequency ECRH & LHCD together with off-axis negative-ion neutral beam injection will be used for achieving steady-state advanced 1 GW fusion power operation for the new design. The more detailed design information and new R&D activities will be introduced in the paper.

Further efforts and challenges: CFETR should get financial support for two phases respectively:

- The budget for engineering design and some R&D have been approved;
- To be approved for CFETR construction by government will be a great challenge and further significant efforts will be needed.
- Wide international exchanges and collaborations will be promoted and welcome !



Figure 1: Roadmap of CN MFE research

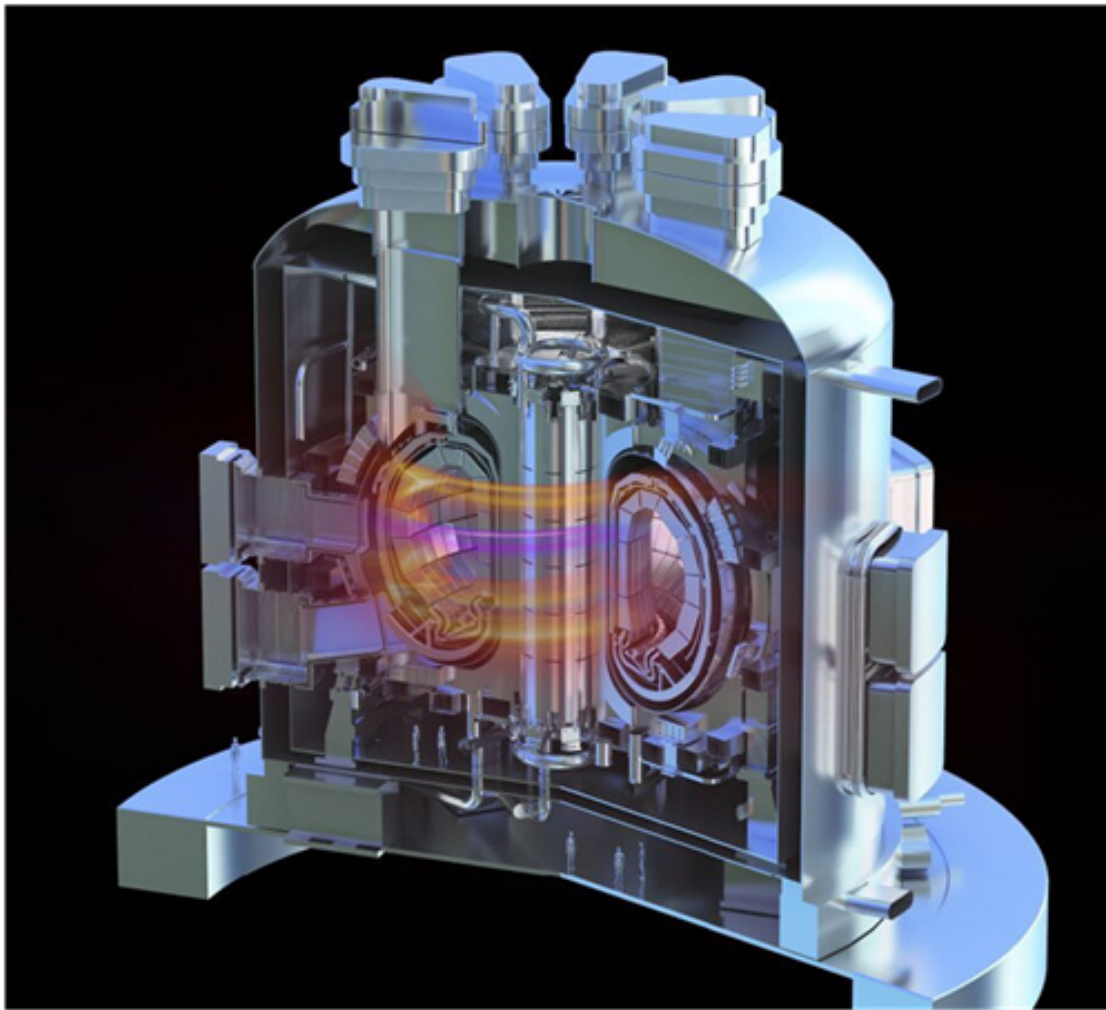


Figure 2: First design version of CFETR

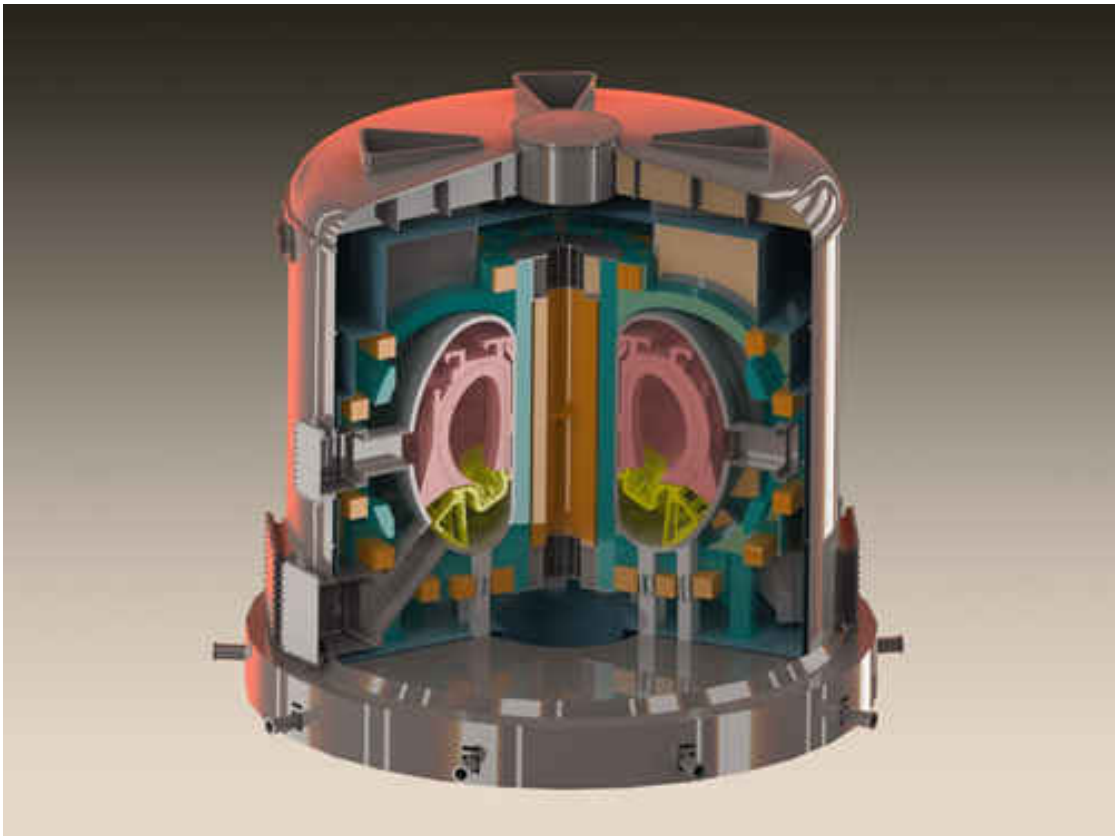


Figure 3: New design version of CFETR

Reference:

1 Wan Y.X, Li J, Liu Y, Wang X.L and CFETR team "Overview of the Present Progresses and Activities on the Chinese Fusion Engineering Test Reactor" 26th IAEA 2016 FEC OV-3

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Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 95 / 303

Design of a high power and low parasitic inductance resistor

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A current-pulsed power supply (CPPS) with rapid rising and falling edges, which is used for tearing mode (TM) feedback control, has been developed for magnetic perturbation coils on the J-TEXT tokamak. A bleeder resistor ranging from 200 m Ω to 1000 m Ω is required in the CPPS. When CPPS works regularly in 0.3 seconds with frequency ranging from 1 kHz to 3 kHz, the bleeder resistor will generate heat which can reach tens of thousands of Joules. And instantaneous power of the bleeder resistor can also reach several megawatts. Besides, in order to reduce the spike voltage during the

insulate gate bipolar transistor (IGBT) turn-off switching transient in CPPS, the low parasitic inductance of the bleeder resistor is required.

Through the above description, if the wire wound resistor is considered as the bleeder resistor in CPPS, the wire with smaller cross section or longer length will be adopted in order to meet resistance value. However, the wire with smaller cross section will be fused by high instantaneous power. Or the longer wire will have more parasitic inductance. The application of a graphite resistor can avoid these problems. On one hand, the graphite with high temperature resistance, is suitable for high power applications. On the other hand, the resistivity of special treated graphite can reach $1.7 \times 10^{-5} \Omega \cdot m$, which is about 1000 times that of copper conductor. Hence, the graphite resistor can be designed with a shorter length and larger cross section. In order to further reduce the parasitic inductance, the graphite resistor is divided into several graphite rods which are placed side by side and connected in series. The parasitic inductances in different layouts of graphite resistor are analyzed through the simulation, and at last, the two-rows and six-columns layout is adopted. The test result of parasitic inductance is about 242 nH, which is agree with the simulation result. Finally, graphite resistor is applied in CPPS and experiment results verified the effectiveness of the designed graphite resistor. Meanwhile, the graphite resistor with low parasitic inductance can be also applied in other high power occasions.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 110 / 305

Shutdown Dose Rate Calculation for the Preliminary Concept of K-DEMO Equatorial Port Area

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The Korean fusion demonstration reactor (K-DEMO) will be operated in a highly irradiated condition by 14 MeV neutrons from D-T plasma. During this condition, irradiated materials generate radioactive nuclides. The nuclides emit decay gammas during operation and even after the shutdown of the tokamak reactor. One of the important safety-related maintenance areas in the tokamak reactor is the outboard equatorial port area. Although it is close to highly irradiated plasma facing components, the human access is necessary for the maintenance. Thus, the reliable result for the shutdown dose rate calculation has to be presented to assure the human safety. The preliminary concept of K-DEMO equatorial port was developed and then, it was transported into the K-DEMO neutronic analysis model 1. This model adopted the labyrinth structure to prevent neutron leakages between the equatorial port structure and neighboring components. The shutdown dose rate calculations have been performed in the vicinity of the equatorial port area based on the rigorous 2-step (R2S) method [2]. This method couples transport and activation codes of the MCNP [3] and FISPACT [4]. The shielding calculation by changing shield thickness has also been performed to provide adequate neutron and radiation shields to reduce the dose level at the equatorial port interspace. The preliminary analysis results indicate that the dose level in this area is below the design target value of 100 $\mu\text{Sv/h}$ at 12 days after shutdown.

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Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 73 / 306

Control system designed for the electron cyclotron resonance heating system on J-TEXT

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A control system based on compact reconfigurable I/O (cRIO)-9068 platform of National Instruments has been designed for the electron cyclotron resonance heating system on J-TEXT. The control system is mainly used for monitoring, timing, fast protection and slow protection of the ECRH system. The response time of fast protection is less than 10 μ s based on voltage comparators and field programmable gate array integrated on cRIO, and all the signals are transmitted with the fiber for isolation and stability transmission in the control system. The test results indicate that the designed control system can meet the requirements of the electron cyclotron resonance heating system of the J-TEXT.

Eligible for student paper award?:

No

W.OP3: Blankets and Tritium Breeding: Solid Breeders / 307

Parametric analysis of the EU DEMO HCPB breeding blanket thermal-hydraulic transient operation using the GETTHEM code

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In the frame of the EU DEMO design and analysis activities, the development of a system-level thermal-hydraulic model (the GEneral Tokamak THErmal-hydraulic Model –GETTHEM) of the EU DEMO tokamak has been recently launched at Politecnico di Torino, with the aim of building a tool which would allow a reasonably fast simulation of the entire power conversion system. The

GETTHEM development focused so far on the Breeding Blanket (BB) cooling loops, as the BB is the component which has to remove the highest fraction of power from the plasma (~80 %). Taking advantage of the modular approach typical of object-oriented programming, the models for the cooling elements of both the Helium-Cooled Pebble Bed (HCPB) and the Water-Cooled Lithium-Lead (WCLL) BB concepts have been already developed and applied to the analysis of a demonstrative scenario. GETTHEM has so far proved to be very fast, allowing transient simulation of the nominal operation scenario of an entire blanket segment in almost real-time, and it was also used to analyse the evolution of accidental transients, such as an in-vessel Loss of Coolant Accident (LOCA), for both coolant options.

In this paper, we present the first application of the HCPB module of the tool, which has been updated to the most recent HCPB design, to the parametric evaluation of different plasma scenarios. In particular, the effect of the variation of the heat load caused by some plasma transient events, both operating and off-normal (e.g. due to plasma instabilities) on the overall cooling performance of an entire blanket segment is shown and discussed. Particular attention is devoted to the temperature distribution in the EUROFER structure, which should be kept below the design margin of 550 °C.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 18 / 308

REFMULF: 2D Full-wave FDTD Full Polarization Maxwell Code

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An important tool for the progress of reflectometry is numerical simulation, able to assess the measuring capabilities of existing systems and to predict the performance of future ones in machines such as ITER and DEMO. A novel 2D full-wave FDTD code, REFMULF, presented here is able to cope with full polarization waves, coupling the Transverse-Electric Mode (TE, X-mode) with the Transverse-Magnetic Mode (TM, O-mode) via a linear vectorial differential equation for \mathbf{J} with a generic external magnetic field \mathbf{B}_0 . This equation, coupling wave propagation, described by Maxwell curl equations to the plasma media, is solved using a modified Xu-Yuan kernel [1, [2] with extended long-run stability. The external magnetic field components of \mathbf{B}_0 lying on the propagation plane are responsible for linking the TE and TM modes. For a \mathbf{B}_0 purely perpendicular to the propagation plane the code describes simultaneously o-mode and x-mode propagation. This code enlarges the possibilities of simulation of microwave reflectometry, including depolarization processes in turbulent plasmas, offering capabilities unavailable in present day 2D reflectometry codes, closing the gap to the much sought-after computationally affordable 3D code. Being a parallel code is able to cope with real size problems. Although originally written with reflectometry in mind the code can be useful to simulate other diagnostics such as Collective Thomson Scattering, or Electron Cyclotron Resonant Heating.

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Eligible for student paper award?:

No

W.OP1: Magnets / 309**Development and applications of Magnets module for SYCOMORE CEA system code****Author:** Louis ZANI¹**Co-authors:** Cédric Reux ¹; Jean-François Artaud ¹; Jean-Luc Duchateau ¹; Patrick Hertout ¹; Arthur Boutry ¹; Julien Said ¹; Luc Di Gallo ¹¹ CEA-IRFM**Corresponding Authors:** louis.zani@cea.fr, patrick.hertout@cea.fr, jean-francois.artaud@cea.fr, jean-luc.duchateau@cea.fr, cedric.reux@cea.fr

In the framework of prospective activities for a demonstration power plant, DEMO will be the next step for fusion energy following ITER tokamak. Some of the key design top-level questions can be addressed using macroscopic system-level codes. Those system codes aim to model the whole plant with all its subsystems and identify the impact of their interactions on the design choices. The SYCOMORE code is a modular system code developed to address key questions relevant to tokamak fusion reactor design by giving a global view in technology and physic domains. Among all components, SYCOMORE provides a representation of the magnet system, which is of importance regarding some factor of merits, e.g. fusion power or cost. SYCOMORE is ultimately coupled with an optimizer, scanning a high number of operation configurations and ranking them along selected merits. This scanning requires fast computation for each scanned point, so the magnets modelling must meet a trade-off between simplicity and accuracy. In this paper we describe the way Toroidal Field (TF), Central Solenoid (CS) and Poloidal Field (PF) systems modelling was chosen taking into consideration the driving design criteria used in usual magnets design method (temperature margin, copper maximum temperature during quench, mechanical resilience in stainless steel structural parts, etc...). The specificities of the reduced magnet representation chosen approach will be shown with the benchmarking of the simplified model of the two main systems (TF and CS) applied on ITER and DEMO reactors configurations and compared with output of more sophisticated design processes (detailed analyses, finite elements analyses etc...), together with a discussion on the limits of this approach. PF system implementation being in early stage the principles will be exposed. Parametric explorations of DEMO operation domains will be reported along different conceptual choices related to the magnets (e.g. superconductor material and performances, structure resilience limits...) or on system requirements (e.g. burn duration, net electric power...) and the impact on the magnet system main design features will be exposed. In the other way round, an overall discussion will also be led on the machine performances sensitivity to baseline choices for magnet systems. Finally some general recommendations will tentatively be provided.

Eligible for student paper award?:

No

W.OP2: Heating and Current Drive / 310**Design, Test and Analysis of a Gyrotron Cavity Mock-up cooled using Mini-Channels****Author:** Andrea Bertinetti¹**Co-authors:** Etienne Perial ²; Ferran Albajar ³; Francesca Cau ³; François Legrand ²; Guillaume Ritz ⁴; Laura Savoldi ¹; Roberto Zanino ¹

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The cooling enhancement of the water-cooled resonance cavity in the design of the European 170 GHz, 1 MW gyrotron for ITER, where the peak heat flux can reach 20+ MW/m², albeit on a very short (~ 1 cm) region, is currently based on the Raschig Rings technology. As an alternative to that, mini-channels drilled in a copper annulus have been recently proposed, also based on Computational Fluid Dynamics (CFD) analysis, which demonstrated that this solution could reach higher cooling performance than other possible alternatives: the high Reynolds number mainly due to the high fluid speed in the mini-channels, in fact, guarantees a high heat transfer coefficient, with local boiling occurring in the peak load region. However, the CFD model suffers from several uncertainties related to the presence of free parameters in the boiling model, implemented in the commercial tool STAR-CCM+, adopted in the simulations, and an experimental proof of the cooling capability of the mini-channel configuration was needed.

In 2016, a planar mock-up using the mini-channels cooling option, fully relevant for the cavity operating conditions, was first designed, based again on CFD analysis, then built by Thales Electron Devices and eventually tested at the Areva premises at Le Creusot (France).

The mock-up is made of Glidcop®, a copper-based alloy also used in the construction of the full size gyrotron cavity. An electron gun with a 28 mm x 28 mm square shaped footprint was used to heat the target region of the mock-up. The mock-up was equipped with 9 thermocouples anchored at different heights above the heated surface; the test facility was equipped with a pyrometer, pointing at the middle of the mock-up heated surface, and with an infra-red camera, both dedicated to the measurement of the target surface temperature. A flowmeter, and the pressure taps at the mock-up inlet and outlet completed the test setup. The test matrix included tests at four different mass flow rates, exploiting a large range of power density up to ~ 30 MW/m².

In the paper, the test results are first presented and discussed, with particular reference to the reliability of the surface temperature measurements and to the heat removal capability of the mini-channel cooling configuration. Coming to the comparison with the CFD model results, the available dataset at low heat fluxes, where boiling is not present in the simulations, is used to confirm the suitability of the turbulence model adopted in the simulations. A second subset of the experimental data is then used to calibrate the CFD model when the boiling regime is entered, and the model results are validated against the rest of the database at high heat fluxes. This calibration/validation exercise allows obtaining a reliable numerical tool, which will be used for the simulation of the full-size gyrotron cavity operation, to assess predictively the benefits of this cooling option with respect to that currently adopted.

Eligible for student paper award?:

No

T.OP2: Fueling, Exhaust, and Vacuum Systems / 311

Core fueling of DEMO by direct line injection of high-speed pellets from the HFS

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Core fuelling of a DEMO tokamak fusion reactor is under investigation within the EUROfusion Work Package “Tritium, Fuelling and Vacuum” (WP-TFV). An extensive analysis of fuelling requirements and of related presently available fuelling technologies, indicate that pellet injection still represents, to date, the most realistic option. Modelling of both pellet penetration and fuel deposition profiles for different injection locations, for specific DEMO plasma reference scenarios and assuming the ITER reference pellet mass (6×10^{21} atoms), indicates that Low Field Side (LFS) injection is inadequate, while effective core fuelling can be achieved launching pellets from the High Field Side (HFS) at ≈ 1 km/s. Vertical injection may be effective only provided that pellets are injected at relatively high speed from a radial position ≈ 8 m, which may depend, however, on the specific plasma scenario. An innovative approach, aimed at identifying suitable inboard “direct line” (or “free-flight”) paths, to inject high-speed pellets from the HFS, has been recently proposed as a potential backup solution, and is presently undergoing preliminary investigation. The fuel deposition profiles that can be achieved by this approach are being explored using the HP12 simulation code. The results of some preliminary simulations will be presented, showing that there are probably geometrical possibilities for direct line injection that can provide good fuelling performance, despite they do not aim at the plasma core (for fear of an excessive neutron flux across the direct line of sight of the injection path). However, these trajectories are rather peripheral and thus quite sensitive to the plasma scenario, so that high injection speeds are required in this configuration in order to maintain the component of the pellet velocity perpendicular to the flux surfaces at a large enough value. The deposition profile obtained simulating the injection of 3 km/s pellets through a direct line injection path forming an angle of 68° with the equatorial mid-plane, is quite similar to that obtained injecting pellets at 1 km/s from the HFS, by means of curved guide tubes. The problem of neutron flux deserves however further investigation; if neutrons will turn out to not represent a serious issue, constraints on the injection angle could be relaxed, and further simulations could probably yield better fuel deposition profiles. The identification and integration of suitable tilted straight injection paths may be a rather difficult task, due to the many constraints and to interference with existing structures, including the breeding blanket (BB). In this perspective, the angle scatter of high-speed direct flight injection, and/or the suitability of straight guide tubes to reduce the scatter cone and the corresponding open cross section on BB penetration, are of main interest. An experimental program, aimed at solving these technological issues, has been recently started in collaboration with ORNL, using an existing ENEA/ORNL two-stage pneumatic D2 pellet injector. Preliminary activities related to this program will also be reported.

Eligible for student paper award?:

No

W.OA3: Neutronics and Multiphysics Analysis / 312

Design, Research and Development of CFETR Vacuum Vessel

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The vacuum vessel, as one of the important components for the Chinese Fusion Engineering Testing Reactor (CFETR) superconducting magnet Tokamak, can provide ultra-high vacuum and clean environment for the plasma operation. The CFETR vacuum vessel was preliminary designed to be a torus with D-shaped cross-section, 4 upper vertical ports, 8 lower ports and 6 equatorial ports, which will be introduced detailed in the paper firstly. In order to verify the design and key technology to be used in the future, 1/8 VV sector mockup is designed and manufactured in ASIPP now. It was

used to demonstrate the make forming, welding, cutting, NDT processing and all kind of tools technology and development. Main design parameter and characteristic of the 1/8 VV sector mockup was introduced secondly, including the inner shells, outer shells and stiffening ribs between them, straight line, arcs and the tangential joint between them. As the results of calculating thermal Analysis, structural analyses under the combined loads of gravity, electromagnetic and neutron thermal radiation, stress and deformation on CFETR VV can be obtained, which is useful for the structural design of VV. At last the Research and Development (R&D) of key technologies to the VV manufacture which have been carried out in ASIPP and the assembly technology for the 1/32 VV sector mockup were discussed and introduced.

Keywords: CFETR vacuum vessel; structural analysis; Assembly sequence; Research and Development (R&D)

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 47 / 313

Qualification of ITER PF6 Helium Inlet

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The Poloidal Field (PF) coils are one of the main sub-systems of the ITER magnets. The PF6 coil is being manufactured by the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP) as per the Poloidal Field coils cooperation agreement signed between ASIPP and Fusion for Energy (F4E). The ITER PF6 winding pack is composed by stacking of 9 double pancakes. Each double pancake is supplied with supercritical helium with 2 inlets located on the innermost turn, in the middle of the layer joggles where the conductor goes from a pancake to the other. The helium inlet will undergo huge cyclic electromagnet loads during Tokamak operation, thus needs to be qualified with rigorous procedures.

This paper describes PF6 helium inlet qualification. In the qualification process, helium inlet hole drilling and stainless steel wrapping removal were carried out. Helium inlet welding with full penetration by automatic welding machine was performed, temperature measurement during welding was implemented and was under 250 °C. Leak test and X-ray test were applied to ensure no defect was exist. The qualification samples passed the 600,000 cycles fatigue test and laminography test. Micro and macro inspections were been done to finally check the welding quality. PF6 helium inlet qualification was approved by ITER IO before Dummy double pancake manufacturing.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 114 / 314

Study of fire impact on detritiation of atmosphere in tritium handling facility: catalytic oxidation of fume gas produced by cable burning

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Fire is one of main scenario for accidental tritium release in tritium handling facilities. Preventing this tritium escape to the environment requires maintaining sub-atmospheric pressure in the affected rooms and detritiation of the gas prior to its discharge. In all gas detritiation systems designed to process a large gas flow (in JET, ITER) first operation stage is catalytic conversion of tritium in hydrogen-containing gases to form of tritiated water. Then the tritiated water is either removed from the gas steam by its drying or detritiated by phase isotopic exchange with liquid water. For handling accidental tritium leak to atmosphere of the room affected by fire the challenge is to ensure operability and efficiency of catalytic recombiner. Gaseous hydrocarbons unavoidably produced during fire present a source of fuel for catalytic recombiner. Their oxidation in exothermic reactions will result in rise of the catalyst's temperature. Because power supply and I&C cables are most common fire load this study focused on catalyst behavior in oxidation of fume gas produced in cable's burning in air atmosphere. Cables for power supply and I&C of low-halogen ALSECURE type from NEXANS S.A., France were used in experimental tests. Their burning was performed in electrical furnace under purge with constant flow of ambient air.

An analysis of the flue gas's composition showed that the combustion of polymeric materials in cable's electrical insulation occurs under oxygen starvation conditions. The flue gas contains a large number of different organic products of insulation's thermal cracking in addition to carbon monoxide. Aerosols were removed from the gas stream by its filtration through HEPA filter. Prior to injection to the catalytic recombiner the gas stream was mixed with an additional flow of atmospheric air and heated to operation temperature of the recombiner (473K). Mixing of gas stream from the furnace with an additional air stream was at several ratio, 1:3.5, 1:8, 1:27 and 1:80. Temperature of of gas stream at recombiner's inlet and catalyst's temperature at various points of the recombiner were measured continuously. The recombiner was filled with catalyst which contains 0.5weight % of platinum on alumina.

It was observed that increase of catalyst temperature depends on mixing ration of the gas from furnace with stream of additional air. For example, at ratio 1:3.5 catalyst temperature rised from 473K to reacted 1570K and the gas's temperature from 473K to 1270K. At ration 1:8 highest catalyst's temperature exceeds upper limit for the thermocouples. Recombiner investigation after this test reveals that internal components made of stainless steel were melted down. With further increase of the mixing ratio to 1:27 rise of catalyst's temperature felled down to 970K. At mixing ratio of 1:80 no temperature rise was detected.

The experimental data were compared with mathematical modeling of the process. Heat transfer parameters of the recombiner needed for the model were evaluated by measuring temperatures rise in tests with air containing constant concentration of hydrogen in air. Comparison of the experimentally measured and calculated temperatures indicates a satisfactory adequacy of the model for the process interpretation.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 19 / 315

Evaluation of the distribution of C5+ and Li2+ by the VUV imaging system on EAST

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The Chinese Fusion Engineering Test Reactor (CFETR) is designed as the next fusion device in China aiming to bridge the gaps between the fusion experimental reactor ITER and the demonstration

reactor (DEMO). The current EAST tokamak will provide a long-pulse, high power test bench for advanced operation scenarios under actively cooled metal wall condition for CFETR 1. To achieve long-pulse and high power steady state operation on EAST, impurity accumulation is one of the key issues should be considered. Therefore, studies on impurity transport become important.

Carbon is one of the major intrinsic impurities in EAST. Additionally, Lithium may exist through the dedicated Lithium-related experiment by Li-pellet injection in the experiment or through wall conditioning by lithiation. On EAST, a new vacuum ultraviolet (VUV) imaging system is developing. It selectively measures the emission with 13.5 nm in wavelength, which is mainly contributed by C VI ($n = 4-2$ transition), or the Ly- α line emission from Li III on EAST. In this work, a new method is proposed to evaluate the distribution of C5+ and Li2+ from the VUV imaging data. In this proposal, the evolutions of the impurity profiles are calculated by the 1D transport model. With the measured electron temperature and density profiles, the emissivity can be estimated. Then, the VUV images can be simulated. Finally, the impurity distributions can be obtained by fitting the simulated images and experimental images.

Acknowledgments

This work is supported by the Natural Science Foundation of China under Contract No. 11605244 and the National Magnetic Confinement Fusion Science Program of China under Contract No. 2014GB106000, 2014GB106001 and 2013GB106000.

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Eligible for student paper award?:

Yes

W.OA3: Neutronics and Multiphysics Analysis / 317

Nuclear and Thermal Analysis of a Reflectometry Diagnostics Concept for DEMO

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Co-authors: Rúben Moutinho¹; Paulo Quental¹; Hugo Policarpo¹; André Lopes¹; António Silva¹; Artur Malaquias¹

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The reflectometry diagnostic (RF) for DEMO is envisaged to provide the 2D electron density profile and to be used as a control diagnostic for the real-time plasma position and shape controller. An initial conceptual study has been done defining the position of the antennas and the routing of the waveguides. In the present design, the integration has been driven by remote handling and blanket interfaces. In order to progress with the system integration, neutronic simulations were performed to find out the cooling requirements. This paper presents the initial nuclear and thermal analyses for the initial design of the reflectometry diagnostic for DEMO. The neutronics simulations were performed using the Monte Carlo simulation program MCNP6 and FENDL 2.1 cross-sections. In a first stage, a CAD model of a 200-mm thick section, divided into 17 sectors (to conform with the blanket configuration) was developed, featuring 40 antennas and the corresponding waveguides. This model was simplified using ANSYS SpaceClaim, to make it suitable for the neutronics simulations, and converted to the MCNP input format using the CAD-based modelling program MCAM. The diagnostics section was then integrated into the 2015 DEMO HCLL MCNP model and filled with the homogeneous breeder blanket mixture used in the HCLL breeding blankets. Simulation results show that the nuclear heat loads reach 7 W/cm³ at the surface of the diagnostics section and at the tips of some antennas. Without an active cooling system, the operating temperatures of the components

under such heat loads would be well above the acceptable from the thermo-mechanical point of view. The thermal analysis presented in this work provides the temperature distribution in the components when subjected both to neutron/gamma irradiation and thermal radiation from the plasma, after the implementation of a preliminary design of the active cooling system. This analysis, based on the CAD models developed for the neutronics simulations, is performed using the commercial finite element software ANSYS v18 Mechanical. The effects of the operating temperatures on the material properties of the plasma-facing, horn-shaped antennas, as well as the performance of the whole diagnostics system from the neutron shielding point of view, are analysed and discussed in this paper.

Eligible for student paper award?:

No

T.OA2: Divertors and PFCs: Tungsten / 318

Thermal Stress Evaluation on the Optimized Shaping Design for Tungsten Monoblock in EAST Divertor

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Abstract—This paper investigates the issue of leading edge for EAST tungsten divertor monoblock which is also concerned in ITER project. Besides the positive effects like reduced risk of cracking, the castellation will lead to the increased probability of melting of the castellated divertor due to local power load on leading edge of the gap. That may introduce unacceptable amount of impurity into plasma and cause damage to the plasma facing components. The chamfering on monoblock is applied in order to avoid melting due to the local heating at leading edge. The previous research [1] had calculated the temperature distribution by employing finite element method and proposed an optimized chamfering geometry for the W monoblock in EAST, which can effectively reduce the maximum temperature under 10 MW/m² heat load. In this work the stress of monoblock under the thermal load is further analyzed by means of finite element software ANSYS in order to evaluate the integrality and lifetime of monoblock. Both of the steady state and transient (e.g. ELM) thermal load are considered in the numerical calculation. According to the results of recent researches [2-4] the cosine law is applied in the calculation for steady state thermal load, and the ion orbit model is used for ELM condition. The behaviour of crack initialization is analyzed by using damage parameter curve which is obtained under creep and fatigue load. Moreover, crack propagation rate model is employed in the fatigue and creep lifetime evaluation for both shaped and unshaped monoblock. The result shows that the shaping for monoblock will lead to movement of the location of highest temperature toward the central region of top surface of monoblock. That will increase the gradient of temperature, so as to enhance the stress at W/Cu interface. The shaping for monoblock can reduce the risk of melting due to overheating on leading edge. However, the life time of monoblock could be shortened due to increased stress.

Keywords—W/Cu monoblock, leading edge, stress, lifetime

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Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 96 / 319

Design of the Alfvén Eigenmodes excitation power supply on J-TEXT

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Alfvén Eigenmodes are instabilities which are considered to be excited by high-energy particles in tokamak plasma. In the future fusion reactor, Alfvén Eigenmodes may change the distribution and transportation of alpha particles. Therefore, the study of Alfvén Eigenmodes is very significant. In present tokamaks with relatively low parameters, external antennas are often used to excite the Alfvén Eigenmodes. The J-TEXT tokamak device is also equipped with the corresponding excitation coil. Under the typical operating conditions of J-TEXT, the excitation frequency needs to be in the range of 200 kHz to 700 kHz and the current needs 30A. At the same time, the frequency of the excitation signal is also required to be automatically scanned once time in 300ms. Based on the above requirements, we have completed the design of the excitation power supply.

The main circuit includes uncontrolled rectifier, buck converter and MOSFET full-bridge series resonance inverter. In order to achieve the resonant frequency of automatic scanning, on one hand, the resonant capacitor is adjusted on-line; on the other hand, the controller automatically tracks the resonant frequency. In the implementation process, we divide the whole frequency range into several pieces. For each one of them, we need to match different resonant capacitor. Specifically, the resonant capacitor is arranged in the form of a fixed capacitor in parallel with an adjustable capacitor. The adjustable capacitor we used is a vacuum ceramic capacitor. When one of the electrodes is rotated, the effective area of the capacitor electrodes can be adjusted, so as to change the capacitance value. In design, the rotation of the capacitor electrode is driven by a brushless DC motor, with a corresponding and continuous change of the capacitance.

The inductance of the excitation coil installed on J-TEXT device is 37.5uH. A 1nF adjustable capacitor is applied to the experiment. In the last campaign, we picked up the 2.3nF fixed capacitor so that the resonant capacitance changed from 3.3nF to 2.3nF. Finally, the formal plasma discharging experiments have been carried out on the J-TEXT. The results shows that the excitation frequency could be changed from 450 kHz to 510 kHz in 300ms; the output current can reach 50A. In general, the experimental results verify the feasibility of the design. During the symposium, the detailed design and test results of the power supply will be given.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 64 / 321

STOCHASTIC COST ANALYSIS OF STEADY STATE AND PULSED DEMO-LIKE FUSION POWER PLANTS

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The debate on which will be the operative mode of a future commercial fusion power plant, steady state or pulsed, is still open even in the on going pre-conceptual design phase of the European demonstration power plant, DEMO.

As part of a series of studies on the economics of alternative options of commercial fusion power plant carried out with the FRESCO (Fusion Reactors Simplified Cost) code, the work here presented provides further insights on the issue.

FRESCO is based on simplified models of physics, engineering and economics of a TOKAMAK-like pulsed or steady state fusion power plant. While the model of the BoP (Balance of plant) is derived from the PPCS study and is usually kept fixed, the assumptions on the plasma physics and technological choices can be adapted to the case of study. The assessment of the economics of the power plant is aimed at estimating the effects of specific plant features on the cost of electricity (COE) rather than formulating forecasts. Then, stochastic analyses based on the Monte Carlo method are carried out in order to assess the weight on the COE of uncertainties on multiple aspects (e.g. on the costs of both raw materials and components themselves, on the actual lifetime of plasma facing components and the power plant as a whole, on the power plant financing, and so on).

Two alternative DEMO-like power plants are modelled with FRESCO. Both provide 500MWe power and rely on the same plasma model as the pulsed DEMO1 and the steady state DEMO2, the current European design options. As an extension of the previous analyses carried out with FRESCO, which estimated the effects of the pulse duration and H&CD efficiency on the COE of a pulsed power plant, here the effects of the uncertainties related to each specific operative mode are evaluated through stochastic analyses. For example, the uncertainties on the maximum number of cycles the power plant components can withstand deeply affect the COE range. Discussions on which conditions could lower the COE are provided along with considerations on the related probability of this event to occur.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 20 / 322

Endoscope Emulator Test Stand for ITER Dust Monitor Diagnostic

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ITER is a basic nuclear installation and as such, safety is one of the most important drivers in the design. The ITER Licensing agreement requires that the quantity of dust in the vacuum vessel must

remain below given limits. The maximum amount of mobilizable dust in the vessel is 1000 kg. A technique based on a flexible endoscope was selected as a tool for diagnostic of dust in ITER. The diagnostic will consist of two tools –one for fine viewing of dust with a resolution down to a few tens of microns in few mm spot and another one for dust collection. Both of endoscopes will have coarse viewing with resolution of few hundred microns over wider area to allow the possibility for more general inspection of the surrounding environment. The Endoscope will have to work in a harsh environment where the activation limit reaches a few hundreds of Gy/h, at a magnetic field of about 8T, and at a temperature of 100C. Ideally it will have to work in vacuum in order to allow inspections of the tokamak between shots or after disruptions. The Endoscope will have to go up to 20m deep inside the tokamak to the inspection region. Due to the specific design features of ITER, the endoscope will have to go upward on an inclined surface for inspection about 18 meters away from the insertion point. In order to ensure that the endoscope gets to the desired region of inspection it will be pushed through guide tubes having a number of bends along their length. Initial estimations of endoscope jacket materials, endoscope stiffness and push/pull forces were defined experimentally. This paper will give a brief reminder of the overall strategy for Dust/Erosion/Tritium monitoring in ITER and the role of the dust monitor in this context. It focuses on experimental results of real-size tests inside guide tubes of the behaviour of different endoscope emulators under various conditions.

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Eligible for student paper award?:

No

M.OP2: Materials I / 323

Effects of high-energy C ions irradiation on the D retention behavior in V-5Cr-5Ti

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Alloys based upon the V-Cr-Ti system (e.g. V-4Cr-4Ti, V-5Cr-5Ti) are attractive candidate structural materials in future fusion reactors because of their low activation properties, high thermal stress factor, good strength at elevated temperatures, and usable fabrication properties. However, the high hydrogen isotope retention in vanadium alloys has been a serious concern in the potential application as fusion structural material. In addition, the high fluence radiation of 14 MeV fusion neutrons will produce various kinds of defects, which could make the problem of hydrogen isotope retention in vanadium alloys even worse. So far, this issue has not yet been investigated systematically, due to both the extreme lack of 14 MeV neutron sources and the activation of the neutron irradiated samples. High energy heavy ion beam has long been used to simulate radiation effects of high-energy neutrons. In this paper, samples made of V-5Cr-5Ti alloy are irradiated by 5.5 MeV carbon (C) ions with dose of 2×10^{14} , 1×10^{15} , 3×10^{15} C/cm².

To investigate the defect properties in the irradiated samples, doppler broadening spectrometry of positron annihilation (DBS-PA) tests are carried out at room temperature with an energy-variable slow positron beam. In the measurement, the doppler broadening spectrum of the annihilation radiation was examined by a high-purity Ge detector, recording the gamma rays with energy 499.5–522.5 keV. The S parameter in DBS-PA increases significantly while the W parameter decreases with the increase of the irradiation dose, which indicates that the vacancy-type defects are introduced by C ions irradiation.

To characterize the effects of irradiation on the deuterium (D) retention property of V-5Cr-5Ti, the

irradiated and unirradiated samples are implanted with D in an ECR linear plasma device at the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP). The electron temperature and density were $T_e \approx 2.4$ eV and $n_e \approx 1.5 \times 10^{17}$ m⁻³, respectively. Thermal desorption spectroscopy (TDS) experiments are followed, and the D retention behavior in the irradiated and unirradiated samples are compared and analyzed.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 87 / 324

Study of plasma density effects on the divertor power width of EAST by SOLPS5.0/B2.5-Eirene

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Edge plasma code package SOLPS5.0 is employed to study the effects of upstream density on divertor power width λq for EAST L-mode discharges. The divertor power width, which is an important physical and engineering parameter for diverted tokamak fusion devices, is determined by the parallel and perpendicular transport in the SOL region. Upstream density scan is implemented in the simulation to obtain a wide divertor operational regime from attached divertor regime to detachment. It is found that the divertor power width tends to increase with the increase of plasma density, in consistent with the EAST and multi-machine experimental results. Further analysis shows that the line radiation loss power of CII and CIII in the divertor region move from the far SOL towards the strike point with increasing plasma density. The CII and CIII line radiation dominate the power loss at divertor region. The shift of line radiation loss for CII and CIII may be the main reason for the positive correlation between the edge plasma density and divertor power width. The mechanisms of the changes in carbon impurity radiation and the effects of plasma density on the edge plasma radial and parallel transports will be studied and included in this work to provide a better understanding of the effects of plasma density on divertor power width.

Further work focusing on the effects of other major plasma parameters such as plasma current and heating power on divertor power width will be carried out.

Acknowledgements

This work is supported by National Magnetic Confinement Fusion Science Program of China under contract No. 2014GB11003, 2015GB101003, 2015GB101000; National Nature Science Foundation of China under contract No. 11305206.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 97 / 325

Design of High Precision Power Supply Control System for ITER Platform

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As one of the core equipment of ITER DC steady-state test platform, high precision power supply is a large capacity AC/DC/AC single-phase inverter with current source characteristics, which provides $\pm 2000V / 500A$ output. The difficulties of high precision power supply are high voltage and large current. To meet requirement of high precision output and fast response, the scheme of cascaded H-bridge inverts with IGBT is presented. The switching frequency of high power switching device is low, however low-frequency PWM control will lead to a large number of low frequency harmonics in the output waveform. The technique of carrier phase-shifted PWM based on DSP is applied to decrease the switching loss of IGBT, raise the equal switch-frequency of inverter and improve the performance of output current wave. In the case where the cascaded H-bridges inverter uses unipolar PWM modulation mode, it is verified by MATLAB simulation and mathematical formula that the maximum harmonic is mainly concentrated in the vicinity of $4F \pm 1$ (F is the carrier ratio), the output voltage is 9 levels and the amplitude is 4 times of a single H bridge output voltage. In this paper, the combination of DSP and CPLD are used to obtain sixteen PWM drive signals. First, Using TMS320F2812 on-chip event manager EVA and EVB to get phase complementary drive signal PWM1/PWM2、 PWM7/PWM8. Second, sending the above four drive signals to CPLD, four PWM drive signals with an initial phase angle lag 45° are generated through delaying and phase-shifting. Third, Using sixteen PWM drive signals to control the opening and closing of the IGBT on the four H-bridge arms. Compared with the traditional current closed-loop control strategy, incremental PID control algorithm with anti-integral saturation is adopted in this paper to shorten the time of system stability and improve the accuracy of current. The experimental results show that the control system based on TMS320F2812 can realize the high precision output of power supply.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 107 / 326

Progress on the Design Development for Hard Core Components (HCC) for ITER Diagnostic System

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The hard core component (HCC) is defined for each undesirable situation with cliff-edge effects, defined as:

- 1) Dose to population above 10 mSv
- 2) Contamination of the ground water
- 3) High radiation field which avoies long term human intervention on the site.

Such structure system components (SCCs) are designed to prevent these situations, as well as to return to and maintain a safe state in the event such a situation occurs.

Only 2 hard core situations can really lead to cliff edge effect unless HCC's are implemented to limit the consequences: Extreme Earthquake (SL-3) and multiples fire in tritium building initiated by SL-3.

All penetrations from zone 1 (Gallery) to zone 2 (defined as NBI cell+Port Cell+VVPSS+vault) are HCC's. Either all the penetrations from the external into the building (no matter where) are HCC's. Design Status of the HCC penetrations belong to the USITER Diagnostics System will be presented in this manuscript.

The analysis for the HCC will follow RCC-MR design code (Elastic analysis route) and Level D criterion to demonstrate the integrity and stability of the components for stress test scenario. In case the elastic analysis is found too conservative a limit analysis will be proposed.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 3 / 327

Effect of coolant mass flow rate on flow pulsation in a simplified channel system of CFETR WCCB blanket

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Water cooled ceramic breeder (WCCB) blanket is being developed for China Fusion Engineering Test Reactor. The water with inlet temperature of 285°C and pressure of 15.5MPa is adopted to remove the considerable heat in the blanket, which may boil under the accident condition of loss of coolant. Periodic flow pulsation may be generated in the parallel coolant channels of the blanket. As a result, structural temperature of the blanket may exceed the limit and the structure may even be burned down. Therefore, flow pulsation in the blanket is necessary to be studied from safety point of view. In this contribution, the effect of coolant mass flow rate on the flow pulsation in the WCCB blanket was analyzed by numerical simulation. In order to facilitate comparison with future experiments, the geometric model of a simplified coolant channel system was built. The channel system includes three groups of parallel double channels corresponding to the channels of first wall, cooling plate and stiffening plate, respectively, and four manifolds. The boundary mass flow rate of flow pulsation was obtained. Combined the amplitude and period of flow pulsation with the details of flow in the different kinds of channels, the formation mechanism of flow pulsation was addressed.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 48 / 328

Manufacture and Electrical Properties of Instrumentation Wire Extraction Specimens for the ITER Feeder HV Insulation

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For the quench detection of the superconducting busbars and joints in the ITER Feeder system, several high voltage (HV) instrumentation wires should be led out from the electrical insulation layers to transmit the voltage potential signal to the data acquisition systems. The penetration of the HV instrumentation wires through the intact insulation layers introduces a potential defect in the insulation, which could open up a path from the HV potential to ground; if this were to happen Paschen discharge would inevitably occur, which is the destructive accident for the safe operation of the Feeders.. Since 2015, ITER Organization and ASIPP have collaborated on the architecture and technology of instrumentation wire extraction, and an R&D program was undertaken in ASIPP. In this paper, the detailed HV wire extraction design is presented, along with specimens validating the design. The electrical properties of the wire extractions specimens, including DC hipot test, Paschen test and Partial discharge test, are presented and discussed.

Keywords: high voltage, instrumentation wire, Paschen test, ITER, joint

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 33 / 329

Heat transfer and Structural analyses of a water cooled tube under one-sided heating conditions for fusion reactor divertor

Authors: Liu Ping^{None}; Peng Xuebin^{None}; Song Yuntao^{None}; Mao Xin^{None}

It is necessary to take effective cooling methods and remove the high heat flux from the divertor which sustains one-sided high heat fluxes over 10 MW/m² in steady state. In order to ensure the reliability and security of the divertor in the fusion reactor, it is necessary to conduct the heat transfer and structural assessment of cooling tube in the divertor. In this paper, the heat transfer enhancement of the subcooled flow boiling in vertically upward screw cooling tubes, the tube with twisted tape and plain tube were carried out by using the Fluent software. Then, this paper conducted the structural analyses of the three cooling tube. The ranges of the working parameters of water are as follows: pressure P=4.5 MPa, water temperature at the inlet T_{in}=473 K and mass flux G=8653 kg/m²s. The 3Sm rule, pipe material temperature and the assessment of critical heat flux have been compared with investigated for the three cooling tube, which can guide the optimum for the heat transfer and structural assessment of the monoblock divertor.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 21 / 330

DESIGN OF A HIGH RESOLUTION PROBE (HRP) HEAD FOR ELECTROMAGNETIC TURBULENCE INVESTIGATIONS IN W7-X

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Wendelstein 7-X (W7X) is a large, superconducting stellarator with modular coils and an optimized magnetic field.

A multi-purpose manipulator (MPM) system has been developed and installed on the W7-X vessel, aimed at investigating the edge plasmas of the stellarator. It is a flexible tool for integration of a variety of different diagnostics as e. g. electrical probes, probing magnetic coils, material collection, or material exposition probes, and gas injection. The system is designed as user facility for many diagnostics, which can be mounted on a unique interface without breaking the W7-X vacuum. The

manipulator system, located in the equatorial plane, transports the inserted diagnostic probe to the edge of the inner vacuum vessel. From there the probe can be moved over a maximum distance of 350 mm to different positions inside the plasma with a maximum acceleration and deceleration of 30 m/s².

In the framework of the EUROfusion S1 work program for the preparation and exploitation of W7-X campaigns, a diagnostic insertable probe head called HRP (High Resolution Probe) was developed by Consorzio RFX in collaboration with IPP Greifswald, to study the electrostatic and electromagnetic features of turbulence in the edge region of W7-X using the MPM. In particular the aim of the HRP head is to provide information on parallel current density associated to L-mode filamentary turbulent structures as well as on ELMy structures in H-mode. Furthermore the possibility to measure the time evolution of radial profiles of flow was considered as a further interesting part of the study, given the strong interplay expected between the turbulent fluctuation and the average flows.

The paper reports the design development of the HRP head, from the choice of the sensors to the engineering design. The assumptions and evaluations supporting the main design choices, together with the R&D tests carried out to check the most critical parts, are described in detail.

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

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 66 / 331

Comparison of Deformation Models of Flexible Manipulator Joints for use in DEMO

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A hybrid kinematic manipulator (HKM) is being designed at RACE (Remote Applications in Challenging Environments) to handle the large breeder blanket segments for DEMO. The payload of this HKM is around 80 tonnes, and its trajectory requires stringent position accuracy as it passes key points, in order to manoeuvre the blanket into and out of position in the vacuum vessel. The TARM (Telescopic Articulated Remote Mast) at RACE is also under upgrading, and it is necessary to investigate its deformation displacement due to its massive weight and the payload.

From the past experience of heavy duty robotic machines, it is noticed that deformation of the manipulator joints contribute significantly to the end-effector displacement. In order to compensate such end-effector deformation displacement in the control system, it is necessary to develop computation-effective deformation model of the flexible joints. In addition the deformation model can be further utilized to optimize the end-effector trajectory by using the iterative algorithms.

In order to support the large payload, the joints of the manipulator are complex, making it unreasonable to employ the truss and beam simplifications from the structural mechanics. The finite element analysis (FEA) method can estimate the deformation of a complex structure with high accuracy given the payload, however, its computation consumption makes it prohibitive to apply to the control system and in the iterative algorithms.

The paper proposes two approaches to model the joint deformations: a non-parametric ANN (artificial neural network) model and a parametric model using the Bayesian Markov Monte Carlo method. Both models are trained and identified off-line using a basic dataset from the FEA of the target joints. After the models are well established, they can be used in the control system or iterative optimization algorithms in real-time. In practice, the proposed methods can also be carried out to model the deformation of joints incorporating the transmission mechanisms, based on real on-site measurement data.

The comparative results of applying proposed deformation models on different joints are presented in the paper. The validation of the non-parametric ANN model and the stochastic process based parametric model are conducted, individually, by comparing with the results of applying the FEA on several joints of HKM and TARM. The study can provide a good premise for constructing the entire computation-effective deformation model of manipulators that will be employed in the DEMO.

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

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 22 / 332

Cooling Needs and Thermal Hydraulic Design Studies of Diagnostic Shielding Module of US ITER Port Plugs

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ITER machine will install a set of 45 diagnostics to ensure controlled plasma operation. Many of them are positioned in the upper & equatorial ports. US ITER diagnostics scope includes the design and integration of 2 equatorial port plugs (E03/09) and 2 upper ports (U11/U14). Each port contains three different zones starting from the in-vessel: the port plug zone, the interspace zone and the port cell zone. The diagnostic components in the port plug zone are installed to a large metallic structure assemblies, called diagnostic port plug, consists of three components: Diagnostic First Wall (DFW), Diagnostic Shielding Modules (DSM) and port Plug Structure (PPS). The DFW protects the diagnostic components from plasma neutron and radiation and provides the diagnostic apertures to peer into the plasma. The DSM is designed to support the DFW structures providing neutron shielding together with the DFW. Therefore, the DSM design will cope with the design drive loads from the harsh thermal and electromagnetic environment, especially in the front end. The water channel within the DSM will be designed to allow sufficient cooling during normal operation and for heating during bake-out. The DSMs and its tenant diagnostic systems require the well-distributed balance to limit the maximum temperature range and gradients of various interfaces to ensure the structural integrity. Despite of the challenging design constraints due to various interface requirements, to obtain the optimized cooling water mass flow rates and thermal hydraulic performance will be particularly investigated during the port integration. This paper highlights the study of the cooling needs and thermal hydraulic design for the DSM as one of the design engineering and integration tasks of the US ITER ports.

*This work is supported by US DOE Contract No. DE-AC02-09CH11466. All US activities are managed by the US ITER Project Office, hosted by Oak Ridge National Laboratory with partner labs Princeton Plasma Physics Laboratory and Savannah River National Laboratory. The project is being accomplished through a collaboration of DOE Laboratories, universities and industry. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 88 / 333

R&D of linear plasma facilities for PMI research at ASIPP

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Linear plasma devices can produce low energy, high flux plasmas to simulate boundary conditions in tokamaks. Recently, two new linear plasma facilities have been built at the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP). To understand the hydrogen isotopes behavior in first wall materials, the PREFACE (Permeation and Retention Evaluation FACility for fusion Experiments) machine has been constructed. Another effort is to achieve reactor-relevant divertor plasma parameters in laboratory using RF-based technology. For this purpose, the HPPX (Helicon Physics Prototype eXperiment) machine has been built to address some scientific and technical issues of steady-state helicon plasma discharge.

The main mission of PREFACE is to perform hydrogen plasma-driven permeation (PDP) experiments on plasma facing materials. The PREFACE facility is equipped with a 6 kW@2.45 GHz electron cyclotron resonance (ECR) source and plasmas with a diameter of 40 mm can be produced. The typical electron temperature and density are $T_e = 2-6$ eV and $n_e = 1E16-1E17$ m⁻³, respectively. For PDP experiments, the plasma density should not be too high to avoid the melting of sample membrane. The basic diagnostics includes a Hidden Langmuir probe, an Avantes spectrometer (197-717 nm) and several thermal couples. Hydrogen isotopes permeation and retention data have been taken for materials like tungsten, reduced activation martensitic/ferritic steels and copper alloys in PREFACE. The HPPX facility has a 4 m long vacuum chamber, which consists of four 1 m sectors with an inner diameter of 0.5 m. Modularization design has been applied so that the vacuum vessel can be easily extended for other research purposes in the future. At present, a 13.56/27.12 MHz RF source has been connected to the machine and the maximum power is 50 kW. A steady-state plasma density of $>1E19$ m⁻³ is expected. The electron temperature and density will be further increased by extra plasma heating. An ECR source with a power of over 100 kW has already been proposed.

Eligible for student paper award?:

No

M.OA1: Experimental Devices I / 334

OVERVIEW OF NSTX-U PROGRESS

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The NSTX Upgrade (NSTX-U) team recently completed a scientifically productive research campaign with 10 run weeks of operation. NSTX-U achieved H-mode on the 8th day of operation, surpassed the maximum magnetic field (achieved $B_t = 0.65$ T) and pulse-duration (achieved 2 sec long pulse) of NSTX, matched the best NSTX H-mode performance at ~ 1 MA, identified and corrected dominant error fields, and commissioned all magnetic and kinetic profile diagnostics. In addition to the new centerstack of NSTX-U, which will ultimately double the maximum field and current capability relative to NSTX, NSTX-U also has a more tangential second neutral beam injector (NBI). NSTX-U researchers discovered that this second NBI can suppress Global Alfvén Eigenmodes, which have previously been observed to influence core thermal electron transport. Thus the second NBI may provide means of modifying fast-ion and thermal transport in addition to controlling rotation and current profiles. Finally, NSTX-U researchers implemented new techniques for controlled plasma shut down and disruption detection and commissioned new tools for disruption mitigation. The 2016 run campaign was interrupted by an internal short in a divertor poloidal field coil, and the NSTX-U team is actively developing a recovery strategy. NSTX-U results and future plans will be described

Eligible for student paper award?:

No

W.OP3: Blankets and Tritium Breeding: Solid Breeders / 335

Tritium release from Li_4SiO_4 : The effect of material properties

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Lithium orthosilicate (Li_4SiO_4), in the form of ceramic pebble, is one of the most promising tritium breeder materials for fusion reactor blankets. Particularly, Li_4SiO_4 has been selected as the preferred breeder material for Chinese HCCB-TBM. For efficient extraction and recovery of bred tritium in the breeding blankets, it is important to have a thorough understanding of the mechanisms of tritium release from Li_4SiO_4 ceramic pebbles. In the present study, tritium release behaviors from two batches of Li_4SiO_4 pebbles with different material properties were investigated through out-of-pile experiments. Samples A, which were fabricated by a melt method, had high densities ($\sim 96\%$ TD) and large grain sizes ($100\sim 300$ μm), while samples B, which were fabricated by a wet method, had relatively low densities ($\sim 86\%$ TD) and small grain sizes ($10\sim 50$ μm). The results showed that tritium release temperature from samples A was much higher than that from samples B. Moreover, the fraction of tritium gas released from samples A was much larger than that from samples B, especially under helium purge gas. Based on these observations, the effect of material properties on tritium release behavior from Li_4SiO_4 pebbles was discussed. It was suggested that the grain size played an important role in the tritium release behavior. This study can provide a guideline for optimizing the fabrication process of Li_4SiO_4 pebbles.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 23 / 336

DEVELOPMENT OF RADIATION HARD AND MAGNETIC FIELD COMPATIBLE VACUUM GAUGES FOR THE ITER PROJECT

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The ITER Tokamak, designed to study deuterium-tritium fusion reactions and to demonstrate its viability as a sustainable and clean energy source, is currently being built in South France, on the Cadarache site. Its vacuum system, one of the largest and most complex vacuum systems ever to be built, requires several hundred vacuum sensors for pressure monitoring of its high vacuum systems.

High vacuum gauges operating under magnetic fields as high as 300mT, with gamma radiation in excess of 1MGy and significant neutron fluency, will be necessary in order to fulfill the pressure measurements of the torus vessel, neutral beam injectors, cryostat vessel, diagnostics, cryogenic distribution and heating systems of the ITER Tokamak.

Following an international call for tender and the signature of a Strategic Agreement between ITER and the company INFICON, specific pirani and cold cathode gauges with remote controller have been developed and qualified to operate under the difficult ITER environment.

In this paper the ITER specific environmental conditions and requirements for pressure measurements are reminded. The standardization process for ITER passive vacuum gauges and controllers is then described and emphasis is given toward the products development and qualification testing. Final gauges performances are then detailed and successfully commercialized ITER standard products are lastly exposed.

To complete the picture, highlight is given on additional vacuum sensing development required to complete the ITER vacuum instrumentation portfolio and achieve an operationally safe design.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 58 / 337

Neutronics Analysis of Helium Cooled Ceramic Breeder Blanket with S-shaped Lithium zone and Cooling Pipes for CFETR

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China Fusion Engineering Test Reactor (CFETR) is a tokamak fusion experimental device under design to bridge the R&D gaps between ITER and DEMO. Helium Cooled Ceramic Breeder (HCCB) blanket is one of the candidate blanket concepts for CFETR. Blanket with S-shaped lithium zone and cooling pipes reduces the space of helium manifold and is conducive to tritium breeding. The neutronics analyses of HCCB blanket with S-shaped cooling pipes for CFETR have been performed with the

Monte Carlo code MCNP and nuclear cross-section data from the FENDL-3.1b data library. The aim of the analysis is to provide the reference for the design and optimization of CFETR blanket system. The 3D neutronic analysis for CFETR was done, in which the 11.25 degree sector model (consist of blanket modules, manifold, support plate, shield, divertor, vacuum vessel, thermal shield, TF coils, PF coils, CS and cryostat) was generated with the McCad automated conversion tool from the reference CAD model for analysis, the 2-D (radial and poloidal) neutron source map was plugged via general source definition card to stimulate the D-T fusion neutrons. The concerned neutronics parameters of CFETR, mainly including the tritium breeding ratio to characterize tritium self-sufficiency, the energy multiplication factor to characterize power generation, as well as, the inboard and outboard mid-plane radial profiles of neutron flux densities, helium production rate, displacement damage rate and the energy deposition to characterize the shielding performance, were produced. The detail results will be presented in this conference.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 50 / 338

The deuterium retention behavior in helium irradiated tungsten after plasma exposures in EAST

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Tungsten (W) is considered as the most attractive plasma facing material for future fusion devices attributed to its outstanding properties. During the operation of plasma, helium ions produced by deuterium and tritium fusion reaction will impinge on W, which will raise more complicated tritium retention behavior in W. In the present work, the deuterium retention behavior in helium irradiated W is studied.

Helium ion irradiation experiments with the fluencies of 3×10^{15} , 3×10^{16} and 3×10^{17} ions/cm² have been performed on recrystallized W, respectively. Transmission electron microscope (TEM) observation suggests that large numbers of dislocation loops are generated in W and the size of the dislocation loop increases with irradiation fluence. To understand the deuterium retention behavior in helium irradiated W formed in tokamak environment, the samples are exposed to the EAST tokamak plasma by Material and Plasma Evaluation System (MAPES). The results of thermal desorption spectroscopy (TDS) indicate that the total deuterium retention increase with the irradiation fluence which can be attributed to the defects induced by helium irradiation. The release peak for deuterium at around 403 K and 994 K are observed in all of the samples. Finally, the deuterium behavior in helium irradiated W is discussed in combination with the TEM results.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 83 / 340

GPU parallel Grad-Shafranov solver for real-time equilibrium reconstruction

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To achieve real-time control of tokamak plasmas, the equilibrium reconstruction have to be completed rapidly enough. For EAST experiment case, real-time equilibrium reconstruction is generally required to provide results within 1ms. A GPU parallel Grad-Shafranov solver is developed in P-EFIT code1, which is built with the CUDA™ architecture to takes advantage of massively parallel Graphical Processing Unit(GPU) cores and significantly accelerate the computation. GPU parallel numerical algorithms for block tri-diagonal linear system are implemented based on eigenvalue decomposition and optimized with latest Pascal TITAN X GPU. The solver can complete calculation within 28us with 65×65 grid size and 72us with 129×129 grid size in double floating precision. It supports that P-EFIT can complete one whole equilibrium reconstruction iteration in about 167us with 65×65 grid size and 319us with 129×129 grid size and fulfill the time feasibility for real-time plasma control with both grid sizes. P-EFIT provides a routine real-time plasma equilibrium reconstruction method which has high spatial resolution[2], customized modules and internal current profile calculation for plasma control in EAST.

References

1 Yue, X. N., et al. "Fast equilibrium reconstruction for tokamak discharge control based on GPU." *Plasma Physics and Controlled Fusion* 55.8 (2013): 085016.

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Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 24 / 341

Inertia load analysis of ITER equatorial and upper port plug EPP9 and UPP14

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Inertia load analysis of ITER equatorial and upper port plug EPP9 and UPP14

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The work presented in this paper mainly focuses on the response spectrum analysis of ITER diagnostic equatorial port plug (EPP) and upper port plug (UPP) structure assemblies to extract dynamic behavior of PPs and the in-port diagnostic systems due to transient vacuum vessel (VV) movements during plasma vertical displacement events (VDEs) and seismic loading. The generic port plug structural models were provided by ITER Organization (IO). Based on the generic EPP models, the US

ITER equatorial port #9 Diagnostic Shielding Module (DSM) with in-port systems such as the Electron Cyclotron Emission (ECE) was integrated in and the latest design of closure plate was used to replace the simple plate in the generic model too to ensure structural integrity. For the UPP14 model, diagnostic first wall (DFW), Glow Discharge Cleaning (GDC) system, wide-angle viewing system (WAV) and DSM shielding blocks etc. are updated to latest design based on the generic UPP model. Two types of response spectrum analysis (RSA) were performed: The floor response spectra (FRS) analysis based on random vibration (power spectrum density (PSD)) is to provide the input response spectra for the RSA of next level components in their system integration design and inertial load calculation. The RSA based on deterministic method (Multi-Point Response Spectrum (MPRS)) is to compute the steady state inertial loads of the components currently in our design. Four load cases were simulated and results are provided in this paper: plasma vertical displacement events (VDE-II, VDE-III, VDE-IV) and seismic event (SL-2).

However, RSA gives only steady state result. For VDEs which last only for a very short time, less than half second, the system may not reach steady state so that the energy may not accumulate to reach maximal response as RSA calculated. Thus a transient run would be better to determine the dynamic behavior of the system. There are infinite numbers of time histories that are compatible to a given spectrum. Currently many commercial softwares exist to generate time histories for seismic qualification but almost nothing available for plasma VDEs. This paper provides the typical process to generate artificial time history for VDEs. For our model, two time histories are created and used to run the UPP14 model. Although the two results have relatively big difference at this time, when we have more information on the real disruption behavior of the VV, i.e. how the magnitude gradually increases, holds and decreases etc., this method can be improved.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 27 / 342

Liquid metal natural convection research heat transfer in the presence of a transverse magnetic field

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In the fusion reactor, there are large heat loads in the first wall of facing high temperature plasma to be large temperature differences in the cladding walls to form natural convection and there is a magnetic field to damp out or to stabilize the fluid flow of the liquid metal. Natural convection under a magnetic field is different from the general fluid and in-depth study has important engineering values for the design and safe operation of the fusion reactor.

The MHD buoyant driven convection in a horizontal enclosure of square cross-section has been studied experimentally. In the experiment, we focus on the case where the magnetic field is perpendicular to the applied temperature gradient, and we measure simultaneously the temperature on the plate wall and the fluid velocity by using UDV located on the bottom of the enclosure successfully. The present paper is focused on the influence on the liquid metal heat transfer Nusselt number in the liquid metal with an external, transverse magnetic field. When there is a magnetic field, the velocity and Nusselt number would be strongly influenced. The natural convection flow is from three dimensionality flow to quasi-two-dimensionality flow in the presence of an external magnetic field. The Lorentz force caused by the magnetic field influences the vertical component velocity observably. When the magnetic field is small, the buoyancy force produces the three dimensionality flow including the mainstream clockwise flow and the minor flow in the other plane. If we continue to increase the magnetic field, the electromagnetic force of fluid inhibits the three dimensionality flow. The quasi-two-dimensionality flow would show a biggest velocity in such a case. When there is a strong magnetic field environment, liquid metal natural convection mainstream clockwise flow is inhibited too.

Two different modes are found in different magnetic field strength when there is a magnetic field exerted effects on the liquid metal natural convection. First, when the Stuart number is less than 4, the fluid flow is described as three-dimensional buoyant flow. The convection in the enclosure includes two contributions of inertia, buoyancy and Lorentz force on a quasi-steady flow. Second, when the Stuart number is greater than 4, the fluid flow is the transition from three-dimensional

to quasi-two-dimensional. The inertia can be negligible, and the velocity components of the flow perpendicular to the magnetic field become uniform in the core and exhibit the classical exponential distribution in the Hartmann layers.

The ability of the fitting equation to predict the Nusselt number of the liquid metal natural convection in a strong magnetic field is meaningful in an electromagnetic forces dominated regimes. Two multiple linear regression models of the Nusselt number are summarized which imply that the mechanism of induced current's restraining influence determines the natural convection heat transfer of viscous electric liquids in a strong magnetic field. Research proves that the striking consistency of the magnetic field strong influence on the natural convection heat transfer in different liquid metal and different aspect ratio.

Eligible for student paper award?:

No

W.OP3: Blankets and Tritium Breeding: Solid Breeders / 343

3D Unsteady Model for Be-steam Reaction in Water Cooled Ceramic Breeder Blanket

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Design of the Water Cooled Ceramic Breeder (WCCB) blanket includes beryllium multiplier layers. At elevated temperature, beryllium reacts with steam in an exothermic reaction producing beryllium oxide and hydrogen. Such situation may occur in WCCB in case of the rupture of one of the cooling pipes in the blanket module. This process occurs locally in a complex 3D geometry of the blanket containing several different granular levels and a network of cooling pipes and structural supports. The process is also inherently unsteady since reaction rate depends on concentration of steam and pure beryllium which changes in time. In order to perform detailed analysis of the process the model of the reacting flow through porous media was developed and introduced into 3D CFD code. In this model granular beds are introduced as porous solids simplifying the model geometry, and reducing typical mesh size to manageable amount of tens of millions of elements. Reaction rate between solid beryllium and steam is obtained from experimental results, and depends on temperature and concentration of the reactants. Differential equation for beryllium oxide fraction is introduced, allowing obtaining distributions of beryllium oxide in space and time. Multicomponent flow consisting of a homogenous mixture of steam and hydrogen is considered flowing through the porous solid with variable properties. Sink and source terms for steam and hydrogen fractions are determined by local beryllium oxide mass fraction source according to molar ratios of beryllium steam reaction. Conjugated heat transfer approach is applied to calculate heat transfer in support structures as well as coolant flow, simultaneously with the porous medium steam flow in a blanket's granular beds. The model is validated using experimental data on beryllium steam reaction for granular bed samples.

Eligible for student paper award?:

No

M.OA1: Experimental Devices I / 345

Status of the ITER Vacuum Vessel Manufacturing

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The ITER Vacuum Vessel (VV) has major functions of being the first confinement barrier and removing nuclear heating during fusion reaction of plasma. Also the VV provide mechanical support for all in vessel components such as Blankets, Divertors, In-vessel Coils, Diagnostics, etc. 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 main functions. Therefore in accordance with French regulation the VV and ports are classified as Nuclear Pressure Equipment due to presence of radioactive products in the plasma chamber and in water cooled structure. The VV procurements consist of five Procurement Arrangements (PAs) and four direct investments. The PAs have been signed for the fabrication of nine sectors (five sectors by the EU Domestic Agency (DA) and four sectors by the KO DA), IWS (IN DA), upper ports (RF DA), and equatorial & lower ports (KO DA) in 2008 to 2009. These direct investments are Field joint welding, Instrumentations, In service inspections, and Bellows.

Manufacturing design of VV regular sectors and upper/lower ports have been completed by industries with accommodation of requirements of the RCC-MR 2007 edition and approved by the VV project team and the Agreed Notified Body (ANB). The EU VV sectors are being manufactured by the EU DA with the consortium of Ansaldo, Mangiarotti, and Walter Tosto (AMW). Progress of poloidal segments of the first sector, Sector #5, is about 20 % and other sectors are progressing for manufacturing. The KO VV sectors are also being manufactured by the KO DA with the Hyundai Heavy Industry and progress is about 55 % for the first sector, Sector #6, and about 23 % for the second sector, Sector #1. IN DA with Avasarala Technologies Limited has completed manufacturing of In Wall Shield (IWS) of amount for about 2.5 sectors out of 9 sectors. Remaining of IWS is being manufacturing in order to complete it in end of 2018. Manufacturing of the first upper port stub extension of RF DA with MAN Diesel & Turbo has been completed and full factory acceptance tests have been completed under inspection of ITER organization (IO), related DAs and the ANB. All related manufacturing dossiers have been reviewed by the IO, related DAs/industries, and the ANB under established procedures. Other components for direct investment are under manufacturing design or procurements according to their planed schedule that will be introduced during presentation.

In this report, current progress of manufacturing, intermediate manufacturing results, major difficulties/issues with solutions, and future plan will be presented.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 89 / 346

Development of Neural-Network Potentials for Atomistic Modelling of PWI

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Artificial neural-networks (NN) have been used to model the potential-energy surfaces of, for example, bulk silicon or copper surfaces. NNs may reach the structural and energetic quality of density-functional theory (DFT) at a small computational cost [2]. The authors intend to develop NN potentials based on ab initio data as an alternative approach to empirical potentials for the atomistic modeling of plasma wall interaction processes (PWI).

At the stage of the 'training' process, we obtain data for Be-W surfaces and small molecular clusters of Be_nW_m , Be_nH_m , W_nH_m with $n + m \leq 4$ including also the pure species with $m = 0$.

In the present contribution, we focus on a comparison of quantum chemical methods for Be_nW_m , Be_nH_m , W_nH_m species. Second order perturbation theory (MP2) and coupled cluster CCSD(T) theory are compared with plane-wave and atomic orbital DFT calculations, and with available experimental data. The plane-wave calculations were carried out with the VASP code using the PBE functional for the description of exchange and correlation of valence electrons, and the projector augmented wave approximation for inner shell electrons. The atomic orbital calculations have been performed with the Gaussian code, where we employ the double hybrid B2PLYPD3 and the dispersion corrected $\omega\text{B97X-D}$ functional and also compare the performance of various ECP basis sets. With preliminary results and the knowledge from literature, we can already conclude that a single method will probably not be sufficient to deal with molecular clusters and surfaces alike.

1 Jorg Behler and Michele Parrinello, PRL 98, 146401 (2007)

[2] Nongnuch A., Behler J, PHYSICAL REVIEW B 85, 045439 (2012)

Eligible for student paper award?:

Yes

S.PLN: Plenary S / 347

Status and Plans on MAST-U

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MAST Upgrade, the centre piece of the UK fusion research programme, is expected to be operational by the end of 2017. Its three primary objectives are: 1) development of novel exhaust concepts, 2) contribution to the knowledge base for ITER and 3) to examine the feasibility of the spherical tokamak as a fusion Component Test Facility.

As with most tokamaks, the design and construction of the MAST Upgrade facility has thrown up many engineering challenges, some predictable, some unforeseen. A number of these will be presented including the design of large modular structures for assembly to tight tolerances; methodologies for high-precision magnetic alignment of the 20+ PF coils; the design of a highly flexible divertor with a wide range of diagnostics, configurations and strike point locations; and the extension of the operating limits of copper coil technology to maximise plasma current, TF field and pulse length.

As well as the design challenges, numerous project management and systems engineering lessons have been learned in the fields of planning, estimation, team structure, requirements management, configuration control and many others. Some of the most useful and relevant of these lessons will be shared in the hope that other projects may benefit from findings in common areas of interest.

Finally, with a first campaign now imminent, a summary of some of the near term scientific goals, unique diagnostics capabilities and research opportunities will also be presented.

This work was funded by the RCUK Energy Programme under grant EP/I501045.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 90 / 348

Simulation of turbulent plasma heat flux to the DEMO first wall

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First wall (FW) of the DEMO reactor should protect the breeding blanket and mechanical construction elements of the burning plasma exposure. The plasma impacts the wall surface by heating and by energetic particles. The heat load on FW during steady state burning mainly consists of the following factors: (i) the plasma photonic radiation, (ii) the plasma heat flux along the magnetic field lines, (iii) charge-exchange neutrals, (iv) alpha particles produced by fusion reaction and partially leaked into the scrape-off layer (SOL).

Assessment of the FW heat load is one of the key design issues determining the DEMO reactor, because the heat flux there is a challenge for the FW armor material both due to high operation temperature and considerable erosion rate by sputtering. Cooling system of the first wall in DEMO should provide stable operation in the wide range of surface heat fluxes: from 0.3 MW/m² up to 3-5 MW/m².

In this paper a model for the FW heat load caused by the plasma turbulent heat flux associated with plasma blobs is developed. Plasma blobs, or plasma blob filaments, are localized regions of isolated enhanced plasma density and temperature of a few cm cross-field sizes, spanned along magnetic field in SOL from wall to wall. The blobs propagate in radial direction from separatrix to the wall with rather large velocity, depositing heat flux by electron thermoconductivity and by ion convection parallel to the magnetic field at the intersections with the wall. A model of the blobs moving with constant radial velocity and depositing heat flux to the wall due to these processes has been developed and implemented into the TOKES code, developed over the past decade at FZK-KIT for integrated 2D simulations of transient events in tokamaks.

First simulations of the intermittent turbulent plasma wall heat load due to the blobs for the DEMO-I tokamak reactor design have been performed. Poloidal profile for the heat flux to the toroidally symmetric first wall has been calculated.

Eligible for student paper award?:

No

T.OP2: Fueling, Exhaust, and Vacuum Systems / 349

SUB-DIVERTOR NEUTRAL GAS DYNAMICS: INTEGRATION BETWEEN THE VACUUM SYSTEM AND THE DIVERTOR OPERATION

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Over the last few years much effort has been invested in modeling the complex geometry of divertor and subdivertor region in tokamak fusion devices. The main goal is the investigation of the impact of neutral gas dynamics on the particle removal process, during operation. Depending on the plasma conditions at the divertor, the exhausted neutral gas flow is more likely to be in the continuum and slip regimes near the private-flux region and close to the divertor targets, where neutral-neutral interactions do play a vital role in the flow behavior, and then covers the transitional regime and even the free molecular (collisionless) regime in the subdivertor area. Consequently, a reliable estimate of the macroscopic parameters in such a complex system requires a tool to describe the flow in the whole range of gas rarefaction.

In the field of vacuum gas dynamics the most well-known and reliable numerical tool which is capable of simulating neutral gas flows in the whole range of gas rarefaction is the Direct Simulation Monte Carlo (DSMC) method. In this method, the solution of the Boltzmann kinetic equation is circumvented by simulating the collisions and the ballistic flight of model particles, which statistically mimic the behavior of real molecules.

Recently, a numerical tool called DIVGAS (Divertor Gas Simulator) has been developed at Karlsruhe Institute of Technology (KIT). The DIVGAS code is based on the DSMC method and is mainly focused on fusion applications. For this purpose, the DIVGAS code has been successfully validated using JET experimental results and additionally it has been benchmarked with the neutral code NEUT2D in the JT60SA sub-divertor. Both numerical investigations will be presented as indicative examples.

In view of DEMO, the DIVGAS code will be the main modelling tool, for identifying the design space of an optimum divertor geometry, which features high pumping efficiency. Representative simulations, which have been recently conducted for investigating the influence of the divertor dome on the divertor pumping efficiency, will be demonstrated.

Based on all the above, the present work describes the work flow which has to be set up by applying the DIVGAS code and in parallel highlights how DIVGAS can be successfully applied in modelling a complex sub-divertor geometry in a tokamak fusion device. Our aim is the DIVGAS code to be exploited in the sub-divertor modelling, by illustrating the use of the pumping system as an additional actuator for plasma control and divertor performance optimization.

Eligible for student paper award?:

No

T.OA3: Blankets and Tritium Breeding: Liquid Breeders / 350

Development, characterization and testing of a SiC-based material for Flow Channel Inserts in high temperature DCLL blankets

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Flow Channel Inserts (FCIs) are one of the key elements in the high temperature DCLL blanket concept, one of those being considered for DEMO. FCIs must provide the required thermal insulation between the blanket steel structure and the hot liquid PbLi that is flowing inside them; the high PbLi temperatures (up to 700 °C) allow a high reactor efficiency, but impose a considerable thermal gradient across the FCI's walls, generating mechanical stresses that must be supported without damage during the operation time. Besides, they should provide enough electrical insulation to minimize MHD pressure drop, they must be inert in contact with PbLi preventing corrosion damage, and should present low tritium permeation. To develop a suitable FCIs material with these requirements is one of the main challenges in the development of a high temperature DCLL.

In this research, a SiC-based sandwich material is proposed for FCIs, consisting of a porous SiC core covered by a dense CVD-SiC layer. SiC fulfils the operational requirements for FCIs including low activation and degradation by neutrons, and porous SiC is an attractive candidate to obtain a thermally and electrically low conducting structure; to prevent corrosion by PbLi and tritium permeation, a dense SiC coating is applied on the porous material. To produce the porous SiC core of the sandwich, a method consisting of combining the particle size of the starting SiC powder mixture with a carbonaceous sacrificial phase is proposed, being the sacrificial phase removed after sintering by oxidation. In this work, a description of the production method is presented as well as the properties of the resulting porous material after sintering and oxidation, like porosity, microstructure, thermal and electrical conductivity, and flexural strength. By using this technique, a wide range of porous SiC materials with different porosities and thus, conductivities and strength values, can be produced. According to thermomechanical calculations and FEM models, and assuming a SiC dense coating of 200 µm and a porous SiC core of 5 mm, the core material should present a thermal conductivity $\leq 7 \text{ W/m}\cdot\text{K}$ at 700 °C and mechanical strength $> 50 \text{ MPa}$ to ensure the required insulation and mechanical integrity. In this work, a material with porosity near 45% and thickness $\approx 5 \text{ mm}$, thermal conductivity of 7 W/mK and flexural strength about 100 MPa is proposed as porous core. Porous SiC samples covered by a dense CVD SiC layer of $\approx 200 \text{ }\mu\text{m}$ were tested under hot PbLi to study their response against corrosion. A first batch of samples was tested under static PbLi at 700 °C during 1000 h, after which they did not show any sign of corrosion damage. Then, a second batch of samples was tested under dynamic PbLi flowing at velocity near 10 cm/s at 550 °C during 1000 h. A magnetic field of 1.8T was applied during the test to some of the samples to study its possible effects on the corrosion behaviour. Results of all corrosion tests are presented and discussed.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 25 / 351

Diversification of the position sensing instrumentation for the JET neutral beam calorimeters

Author: Peter Blatchford¹**Co-authors:** Simon Warder¹; Julian Hawes¹¹ *Culham Centre for Fusion Energy***Corresponding Author:** peter.blatchford@ukaea.uk

The JET neutral beam injection system incorporates a calorimeter in each beamline, comprising 2 large cooled copper panels (about 2.5m x 1m) instrumented with thermocouples to provide diagnosis of the beam shape and alignment. The panels are rotated out of the beam path to allow the beam to enter the torus; they are inertially cooled and can only sustain full beam power for a fraction of a second, hence it is essential they are fully withdrawn during plasma operation.

Calorimeter position is monitored with in-vacuum micro-switches close to the limits of travel, but these have proved unreliable in the past; furthermore the panels are known to twist due to a combination of bearing friction, water bellows reaction torque and actuation from the top and as a result may not always reach the switches. This has led to periods of operation where the bottom of the panel has unknowingly scraped the edge of the beam and in 2013 this resulted in melting of the edge of one panel and a large water leak.

A procedure has been implemented to check the calorimeter position and thus avoid a repeat of the melting incident; however in 2015 an independent review panel examined NBI reliability and recommended that a diversity of methods should be used to detect the positions of the calorimeters. This paper summarises the methods considered and details the option selected for installation during the 2017 shutdown.

Key constraints on the technology choice were:

- Compatibility with ultra-high vacuum
- Presence of sputtered copper
- High neutron level (particularly during the planned D-T operation) meaning no active electronics close to the sensor
- Magnetic fields during pulsing
- High levels of vibration
- Measurement accuracy better than 5mm

Many technologies were considered, the most promising being:

- Rows of mechanical or magnetically actuated reed switches, to indicate a series of discrete positions
- Bespoke inductive proximity sensor
- A spring element deflected by the calorimeter movement, instrumented with strain gauges

The latter 2 were investigated in more detail through laboratory experiments and both considered suitable, however the spring element was finally selected on the basis of being considered lower risk.

The detailed design of the final sensor is described, along with the laboratory work on an inductive sensor; this technique was only rejected on the basis of requiring more development work and hence presented a higher risk given the limited time available to design and manufacture a sensor. It may have applicability to other in-vacuum position sensing requirements where more development time and resources are available.

This work has been carried out within the framework of the Contract for the Operation of the JET Facilities and has received funding from the European Union's Horizon 2020 research and innovation programme. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Eligible for student paper award?:

No

R.OP5: Experimental Devices II / 352

Extreme ultraviolet spectroscopy applied to study RMP effects on core impurity concentration in EAST

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Impurity control is one of the key issues to improve confinement performance in present fusion research. The application of resonant magnetic perturbations (RMP) has been proposed as a candidate method to reduce impurity concentration in the core region of magnetically-confined plasmas. In order to understand the effects of RMP on impurity transport, a space-resolved extreme ultraviolet (EUV) spectrometer is used to observe the spatial distribution and temporal behavior of core impurity emissions at the Experimental Advanced Superconducting Tokamak (EAST). The space-resolved EUV spectrometer consists of a 30 μm -width entrance slit, a 1 mm-width spatial resolution slit placed in front of the entrance slit, a concave varied line-spacing (VLS) grating and a back-illuminated charge-coupled device (CCD) detector. Impurity emissions passing through the spatial resolution slit and entrance slit are refracted on the gold-coated holographic grating (1200 grooves/mm) at an incident angle of 87°, resulting in a wavelength resolution of 0.06 Å and 0.15 Å at wavelengths 35 Å and 200 Å, respectively. The spectra in the wavelength range of $30 \leq \lambda \leq 500$ Å are then recorded by the CCD (2048×2048 pixels with a pixel size of $13.5 \times 13.5 \mu\text{m}^2$) installed behind the grating. The vertical profile of impurity emissions in the range of $0 \leq Z \leq 450$ mm is projected on the CCD using the narrow spatial-resolution slit placed horizontally in the spectrometer. In the present study, line emissions of carbon and iron are observed for C VI (33.7 Å, $2p-1s$) and Fe XXIII (132.9 Å, $1s^2 2s 2p-1s^2 2s^2$), respectively, whereas the unresolved transition array (UTA) of tungsten in the wavelength range of $\lambda = 30-70$ Å is monitored to study tungsten behavior when RMP are applied. It is clearly observed that the intensities of C VI, Fe XXIII and W-UTA emissions are reduced by the application of either static or rotating RMP fields, particularly during stages of edge localized modes (ELMs) mitigation in ~ 20 s long pulse discharges. In addition, the reduction in iron and tungsten emissions with $n=1$ and 2 (n : toroidal mode number) RMP is more significant than that of carbon emissions, i.e., $\sim 50-70\%$ for W-UTA and Fe XXIII, while $\sim 40\%$ for C VI. Considering that the line-averaged electron density n_e only decreases $\sim 20\%$ in the same discharges, the EUV spectrometer measurements indicate that RMP can significantly decrease the impurity concentration in the core region on top of overall particle pump out. The spatial distributions of impurity emissions are also measured for different impurity species using the EUV spectrometer in EAST. The results confirm the feasibility of the space-resolved EUV spectrometer to study the effects of RMP on the core impurity concentration based on the measurement of time evolution and spatial distribution of impurity emissions.

Eligible for student paper award?:

Yes

M.OA3: Inertial Fusion Engineering and Alternate Concepts / 353

Magnetized Target Fusion at General Fusion

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Magnetized Target Fusion (MTF) involves rapidly compressing an initial magnetically confined plasma by $>300\times$ volume compression. If near adiabatic compression is achieved, the final plasma the plasma can be heated to > 10 keV, and confined inertially to produce interesting fusion energy gain. General Fusion is developing a compression system using pneumatic pistons to collapse a cavity in formed in liquid lead-lithium, heating a plasma target such as a spheromak or spherical toroid trapped in

the cavity. With a low-cost driver, straightforward heat extraction, good tritium breeding ratio and excellent neutron protection, the concept is promising as a practical power plant. We will review the plasma formation and compression results achieved so far and our plans moving forwards. Work on the compression system will also be described.

Eligible for student paper award?:

No

T.OP1: Power Supply Systems / 354

DESIGN AND MANUFACTURING OF THE SIC-BASED POWER SUPPLY SYSTEM FOR RESISTIVE-WALL-MODE CONTROL IN JT-60SA

Authors: Alberto Ferro¹; Ferdinando Gasparini¹; Elena Gaio¹

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One of the main objectives of JT-60SA, the satellite tokamak under construction in Naka (Japan), is to confine steady-state high-beta plasmas. To reach the desired plasma performance, the control of the instabilities called Resistive Wall Modes (RWMs) is crucial. At this purpose, besides the combination of an in-vessel passive structure (stabilizing plate) and plasma rotation, a dedicated active control system based on 18 in-vessel sector coils has been devised. These coils are placed just behind the tiles of the first wall and around the ports; therefore, due to the plasma proximity and the low shielding effect of the surrounding passive structures, they can efficiently generate fast magnetic fields if properly driven. The strategy is to succeed in controlling RWMs (which grow exponentially) when their amplitude is still low, so that low magnetic field components and relevant current to produce them are sufficient. As a consequence, the power rating of the power supply system is not so demanding, but at the price of high dynamic performance. To achieve an effective control, each coil will be independently fed by a dedicated inverter, rated for 300 A, 240 V, with a closed-loop bandwidth of 3 kHz and a latency between reference and output voltage lower than 50 μ s. These performances exceed those of standard industrial products and call for the adoption of innovative solutions. To exploit the simple and compact H-bridge topology, a switching frequency higher than that sustainable by standard silicon IGBTs is necessary; the adoption of new power semiconductors based on Silicon-Carbide (SiC), capable to switch at 30 kHz (60 kHz equivalent at the output), and a very fast control system have been the key design choices to satisfy the requirements. To prove the feasibility at reasonable cost of this design, an inverter prototype has been developed in 2014, whose performance was very satisfactory. Therefore, the same inverter design has been confirmed also for the final power supply system (called RWM-PS), which is now under manufacturing. This will be composed of an ac disconnecter, a step-down transformer, two ac/dc rectifiers, a distributed dc-link capacitor bank, 18 water-cooled fast inverters and a Local Control Cubicle. Both differential-mode and common-mode filters are foreseen at the inverter outputs; their main aims are to reduce the current ripple and minimize the electro-magnetic interferences due to the very long coaxial cables connecting the fast inverters to the coils. The control system supervises all the plant. Each inverter can be individually controlled in current control loop or in open-loop, the reference being arbitrary and generated by the JT-60SA MHD Controller.

The RWM-PS will be the first power supply system for fast control of plasma instabilities in fusion experiments adopting SiC semiconductors. The paper will give an overview of the final design of the RWM-PS, with particular emphasis on its special features and the solutions adopted to satisfy the critical requirements. The interface issues for an effective integration with the JT-60SA power and control systems will be also treated.

Eligible for student paper award?:

No

T.OP3: Project Management and Systems Engineering / 355

Early lessons from the application of Systems Engineering at UKAEA

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UKAEA have now been applying Systems Engineering across a range of projects for a number of years, both directly and indirectly linked to fusion engineering. Systems Engineering has provided a unique perspective from which to solve these complex engineering challenges, bringing together insights from all aspects and disciplines involved. Fundamental functional requirements of the systems have been captured and used to develop “solution agnostic” designs (or architecture) of each system at the highest functional level. This has yielded two major benefits; Existing preconceptions of the design have been challenged and alternatives assessed against the abstract system architecture to determine the optimal combination of sub-systems. In addition, as the design evolves, it has been possible to check that it is still staying true to the original intent.

Systems Engineering has also provided a rigorous methodology for recording and tracing the system requirements and associated designs down through multiple hierarchical levels with associated acceptance tests. Great care has been taken in the specification of these tests to ensure that when the system is fully commissioned it will adhere to both the fundamental requirements as well as the chosen sub-system design solutions.

This paper will present the lessons learnt and the benefits seen from applying Systems Engineering in a range of projects at UKAEA. It will bring together the work being carried out at UKAEA in applying Systems Engineering to fusion engineering and beyond. It will present case-studies from the European DEMO, both in the overall design and integration of the power plant as well as within specific work packages. It will show how the top-level work has produced a new perspective on the power plant design. In the work packages of remote maintenance & breeder blankets it will discuss how functional preconceptions and assumptions have been challenged leading to improved designs. It will also draw on the experience RACE (UKAEA) have gained from applying Systems Engineering to create an optimised design for the European Spallation Source Active Cells project. Each case study will home in on the aspects of Systems Engineering which have been applied to greatest effect and consider both the short-term benefits already realised and the long-term benefits that are anticipated in the future.

In concluding, it will consider the application of Systems Engineering to the design of fusion power plants, within the wider context of the current trends in the Systems Engineering community and identify possible future avenues for the application of Systems Engineering to fusion.

Eligible for student paper award?:

No

M.OA3: Inertial Fusion Engineering and Alternate Concepts / 356**Experimental results from the SPECTOR device at General Fusion****Authors:** Stephen Howard¹; Michel Laberge¹; Russ Ivanov¹**Co-authors:** Neil Carter¹; Patrick Carle¹; Peter O'Shea¹; Ryan Zindler¹; Aaron Froese¹; Adrian Wong¹; Alex Mossman¹; Curtis Gutjahr¹; Ken Jensen¹; Merritt Reynolds¹¹ *General Fusion***Corresponding Authors:** info@generalfusion.com, stephen.howard@generalfusion.com

General Fusion (GF) is operating a new sequence of plasma devices called SPECTOR (**S**pherical **C**ompact **T**oroid) capable of generating and compressing plasmas with a more spherical form factor, avoiding the concave liner geometry used on previous compression tests at GF. SPECTOR forms spherical tokamak plasmas by coaxial helicity injection into a flux conserver (R= 19 cm, $\lambda_{Taylor} = 23.9 \text{ m}^{-1}$, minor radius of 8.3 cm) with a pre-existing toroidal field created by $\leq 500 \text{ kA}$ of current in an axial shaft. The initial poloidal flux of up to 30 mWb and toroidal plasma current of 100 - 300 kA is formed rapidly in the spherical flux conserver during a Marshall gun discharge (850 kA peak, 90 μs duration), and then resistively decays over a time period of $\sim 2 \text{ ms}$. SPECTOR 1 has an extensive set of plasma diagnostics including a surface magnetic probe array, 3 interferometer chords, visible and VUV spectroscopy, multi-point Thomson scattering as well as a 4-chord FIR polarimeter system in development. SPECTOR 2, 3 are mobile test platforms that can be transported out of the lab for compression tests. Plasma facing surfaces include plasma-sprayed tungsten and bare aluminum, and can be coated with $\sim 5 \mu\text{m}$ of vacuum deposited lithium for the purpose of getting impurities out of the base vacuum and to reduce the gas recycling coefficient of the wall. Working gas has included helium and deuterium. Experimental characterizations have been made of formation dynamics, MHD mode activity, evolution of plasma profiles during its lifetime, and trends in FWHM magnetic lifetime with respect to system control parameters. Control of safety factor profile $q(\Psi)$ can be achieved through a choice of the amount and axial distribution of poloidal gun flux and the amount of shaft current. Grad-Shafranov equilibria are reconstructed from the surface magnetic data using Caltrans/Corsica. Ideal and resistive MHD stability can be tested with DCON and NIMROD over a range of pressure and current profile parameters. Realistic compression scenarios have been simulated using the 3D MHD code VAC. The SPECTOR geometry is stable for a wider range of plasma parameters than previous experiments at GF. Relatively hot ($T_e \geq 400 \text{ eV}$) and dense ($\sim 10^{20} \text{ m}^{-3}$) plasmas have achieved energy confinement times $\tau_E \geq 100 \mu\text{s}$ and are being used in field compression tests.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 51 / 357**Some Properties of Beryllium Pebbles Produced by Powder Metallurgy for HCPB Breeding Blanket Application****Authors:** Igor Kupriyanov¹; G. Nikolaev²; S. Zavjalov²; L. Kurbatova³; N. Zabirowa²; V Chakin⁴¹ *A.A. Bochvar High Technology research Institute of Inorganic Ma*² *A.A. Bochvar Research Institute*

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Beryllium is planned to be used as the neutron multiplier in helium cooled pebble-bed (HCPB) breeding blanket concept for DEMO power plants. Under neutron irradiation a large amount of helium and tritium is produced in beryllium. The key issues of neutron irradiation of beryllium are helium-induced swelling and tritium retention and release. Because of safety requirements, the in-pile tritium release should be sufficiently high to avoid risk to personal in case of a serious accident in a fusion power plant leading to abrupt release of all accumulated tritium.

In the present HCPB breeder blanket design, beryllium is used in the form of pebbles with diameter of ~1 mm, having inherently large grains (in the 500-1000 μm range) due to the fabrication by "Rotation Electrode Method" or "Fluoride Reduction Process". However, it is expected that in beryllium with fine grain structure (average grain size of a few tens micrometers) helium and tritium release can be improved significantly. In order to produce the pebbles with a fine grain structure, some R&D were performed in Bochvar Institute. Several experimental batches of Be pebbles with average pebble size of 1.2 –1.3 mm and different grain sizes (from ~ 13-14 μm up to ~615 μm) have been fabricated by powder metallurgy and then characterized.

This paper presents the results of investigation of three batches of beryllium pebbles with average pebble size of 1.2 –1.3 mm and different average grain sizes (13-14 μm , ~50 μm and ~615 μm). Microstructure and chemical composition of produced beryllium pebbles are presented as well as packing density and pebble size distribution. The influence of grain size on tritium release and retention in Be pebbles during temperature programmed desorption (TPD) after high-temperature loading of tritium/hydrogen gas mixture are also described.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 52 / 358

Flow Test at Factory for ITER Thermal Shield

Authors: Kwanwoo Nam¹; Dongkwon Kang¹; Woocho Chung¹; Kisuk Lim²; Youngkil Kang²

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Thermal Shield (TS) is to be installed between vacuum vessel/cryostat and superconducting magnet in ITER tokamak. The TS plays a role in minimizing thermal radiation load onto the magnet. The TS has to be cooled by flowing 80 K helium gas inside cooling pipes welded on the TS surface. The helium coolant is supplied from the cryoplant via manifold pipes and distributed to all TS segments. Flow through each TS segment should be fully characterized to accurately predict the flow distribution in TS flow network for ensuring reliable operation of the TS.

This paper describes how the manufactured TS segments are validated by factory flow test. Instead of applying cryogenic helium flow in the test, high pressure and room temperature nitrogen gas passes through the cooling pipe on TS segment. Equivalent test flow rate is determined by matching test Reynolds number with that of actual operating condition of TS based on similarity principle. Flow rate is controlled by a thermal mass flow controller and pressure drop between the inlet and the outlet of the pipe routing is measured by a differential pressure gauge. Test results are compared with calculated ones by incompressible pipe flow analysis to check the validity of pipe and elbow loss coefficients for the real manufactured TS segments. Orifice elements are to be connected to several TS segments for mass flow balancing of the TS flow network. The orifices are also tested separately by the flow test apparatus. Correlation for the orifice loss coefficient is derived from the test results.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 10 / 359

Development of I&C main functions for ITER VUV spectrometers and prototype test at KSTAR

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The main I&C (Instrumentation and Control) functions of the ITER VUV (Vacuum Ultraviolet) spectrometer have been prototyped based on ITER CCS(CODAC Core System) and tested at KSTAR. The ITER VUV spectrometer consists of VUV core, divertor VUV and VUV edge, which are expected to use a common VUV detector model, Andor BI(back-illuminated)-CCD(Charge-Coupled Device). While many other auxiliary functions are required for the full plant I&C system, the core functions of the plant I&C system for ITER VUV spectrometer are to acquire VUV spectra from VUV detectors and to trigger the data acquisition at desired timings. A C++ fast controller software as well as an EPICS ioc has been developed on ITER CCS to implement this core function of data acquisition with triggering. Andor Linux SDK(Software Development Kit) was utilized for the implementation of data acquisition from the envisaged VUV detector while NI-SYNC library was utilized for the generation of triggering signal with PXI-6683H timing board. The developed prototype has been tested with the envisaged VUV detector and optical convertor modules at KSTAR. The developed codes will be utilized for the automated VUV data acquisition during 2017 KSTAR campaign.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 11 / 361

Design of the optical emission spectroscopy diagnostic system and preliminary experimental results in RF negative ion source

Authors: zhi-min liu^{None}; yan wang^{None}; jing-yan yan^{None}; li-zhen liang^{None}; chun-dong hu^{None}

The development of radio frequency (RF) negative ion sources for neutral beam systems requires knowledge of the plasma parameters. Optical emission spectroscopy (OES) is a non-invasive and in situ diagnostic tool, so optical emission spectroscopy diagnostic system are designed to be applied to the measurements of the RF negative ion source, and diagnostic principle and simplified analysis methods for plasma parameters are introduced. A preliminary results of a variety of plasma parameters are obtained based on the part of the local thermodynamic equilibrium (PLTE) state. When the

discharge power is 25kW and the discharge operates pressure is 0.5Pa, the electron temperature is about 0.83eV and the positive hydrogen ion density is $2.7 \times 10^{18}/\text{m}^3$.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 12 / 362

Prototype design of a 700 C in-vacuum blackbody source for in-situ calibration of the ITER ECE diagnostic*

Authors: Abdelhamid Ouroua¹; Joseph Beno¹; Adam Bryant¹; Andrei Khodak²; Perry Phillips³; William Rowan³; Gary Taylor²; Damon Weeks¹

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Two blackbody sources permanently located within the ITER Diagnostic Shield Module (DSM) at equatorial port 9 will operate in conjunction with two remotely retractable mirrors to generate and direct blackbody radiation to calibrate the radial and oblique views of the ITER Electron Cyclotron Emission (ECE) diagnostic. The main calibration requirements include a high-emissivity surface heated to 700 C, 5000 hours operational lifetime over 20 years, and ability to perform the calibration in the presence of a magnetic field. Major design drivers include a heater current limit of 40 A, no direct fluid cooling to mitigate failure risks, small size to avoid compromising neutron shielding, and adequate structural support to mitigate vibration loads.

Each calibration source consists of an engineered emissive surface to generate the blackbody emission, a heating element to control the temperature of the emissive surface, heat shields, housing, power cables, and temperature sensors for feedback control. Early R&D focused on the heater and started with a commercial-off-the-shelf Inconel heater. After extensive testing, temperature limits, long term emissivity instability, and material vaporization issues prompted consideration of alternative heaters. A second focus was on efficient transfer of the heat to the emissive surface. When heated from its flat side to 965 C by an open molybdenum wire, a molybdenum plate with machined V-grooves to increase its effective emissivity was found to successfully raise the temperature of the emitter to the required 700 C. Further testing demonstrated long term stability exceeding 120 hours of continuous operation at 700 C and under high vacuum. These results led to a custom encapsulated molybdenum heater designed in close collaboration with a heater manufacturer.

Mechanical support of the silicon carbide emitter presented challenges due to brittleness of ceramics. Thermal expansion, material strength degradation at high temperature, and restrictions on materials allowed in ITER required several design iterations to arrive at an acceptable emitter support solution. Heat management in the hot calibration source also presented challenges because of the goal to avoid direct fluid cooling. While excess heat is allowed to be transferred to the cooled DSM wall, which is separately maintained at 100 C, a calibration source attachment approach that results in adequate heat transfer proved challenging to achieve.

Finally, a closed-loop temperature control approach was developed, implemented, and successfully tested where the emitter temperature was regulated to within 0.5 C.

Description of the preliminary design of the hot calibration source with supporting analyses and test results will be presented in the paper.

*This work is supported by US DOE Contract No. DE-AC02-09CH11466. All US activities are managed by the US ITER Project Office, hosted by Oak Ridge National Laboratory with

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 65 / 363**Advanced shape design with F2EQ code in CFETR****Authors:** Zhengping Luo¹; Bingjia Xiao¹; Jiangang Li¹; Minyou Ye²¹ *Institute of Plasma Physics, Chinese Academy of Sciences*² *School of Nuclear Science and Technology, University of Science and Technology of China***Corresponding Authors:** j_li@ipp.ac.cn, yemy@ustc.edu.cn, zhpluo@ipp.ac.cn, bjxiao@ipp.ac.cn

The Chinese Fusion Engineering Test Reactor (CFETR) is the next device in the roadmap for the realization of fusion energy in China, which aims to bridge the gaps between the fusion experiment ITER and the demonstration reactor (DEMO) 1. CFETR will be operated in two phases: Steady-state operation and self-sufficiency with a modest fusion power of up to 200MW in Phase I, and DEMO validation with 1GW fusion power in Phase II1. A key challenge facing high-power steady-state operations and self-sufficiency is to maintain a fusion plasma with adequate performance while preventing damage to the vessel walls, especially divertors. Except the plasma detachment from a divertor target methods, other solutions that optimize the magnetic field topology in the divertor region have been proposed to reduce the power loads [2,3]. These advanced configurations will be a potential choice for CFETR.

In order to optimize the advanced plasma divertor shape, a new tool dubbed the F2EQ code was developed to fit CFETR shape design. F2EQ is a MATLAB toolbox with a series of scripts and functions, which can be used to solve the fixed-boundary and free-boundary plasma equilibrium problems. It provides the flexibility to set multiple plasma configuration, especially the advanced diverted configuration, like snowflake-plus. Starting from a conventional single null configuration, second order null constraint and multiple first order null constraint on X-points was enforced to obtain the desired advanced plasma shape. In order to satisfy the different concepts and ideals in concept design phase, multiple divertor coils inside or outside the vessel with optimized currents are considered. It is easy to switch any coil on or off with any desired plasma equilibrium in different shape and other parameters, like β and β_p . Based on a series calculation with F2EQ code, the optimized plasma equilibrium with two in-vessel divertor coils with modest current in snowflake-plus configuration gives factor 4 flux expansion then the used case with two external DC coils [4].

1 Y.X. Wan, et al. Overview of the present progress and activities on Chinese Fusion Engineering Test Reactor 26th IAEA Fusion Energy Conference, Kyoto, Japan, 2016 (Submitted to Nuclear. Fusion)

[2] D.D. Ryutov, et al, Phys. Plasmas, 15 (2008) 092501

[3] P.M. Valanju, et al. Phys. Plasmas, 16 (2009) 056110

[4] Z.P. Luo, et al. IEEE Trans. Plasma Sci. 42 (2014) 1021

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 28 / 364**Cooling concepts for CFETR divertor target****Author:** Xuebing PENG¹**Co-authors:** Xiaobo CHANG²; Xin MAO¹; Thomas R. BARRETT³; Minyou YE²; Yuntao SONG¹; Xinyuan QIAN²; Ping LIU¹

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The China Fusion Engineering Testing Reactor (CFETR) aims at bridging the gap between ITER and DEMO. Its scientific mission is to produce fusion power of 200 MW with tritium self-sustention and duty cycle of 0.3-0.5. The big fusion power and the auxiliary heating power of 100-140 MW, makes the design of CFETR divertor challenging. Previous work focuses on the plasma configuration and the first round engineering conceptual design, in which the divertor target employs the ITER-like water-cooled W/Cu monoblock. However, this W/Cu concept is only feasible for the operation phase I when the neutron dose level is comparable with ITER. While in operation phase II, the neutron dose level is much higher, evaluated as 5 dpa/year in the divertor. As a result, the high activation of CuCrZr heatsink prevents the use of W/Cu concept. Therefore, new cooling concepts are being studied.

The first updated one is still based on the W/Cu concept, whereas the CuCrZr is replaced by the China Low Activation Martensitic steel (CLAM). Unfortunately the low thermal conductivity of CLAM, ~28 W/(mK), drastically decreases the heat loads capability. After optimization of geometrical parameters of the monoblock, with proper hydraulic parameters the structure can afford 10 MW/m² heat flux in steady state. In addition, a novel concept was proposed that with tungsten alloy WL10 as heatsink and FLiNaK molten salt as coolant. The initial designed divertor target is a 5 mm thick tungsten tile brazed onto a 1 mm thick filleted rectangle WL10 heatsink. Based on thermo-hydraulic and mechanical calculations, with proper hydraulic parameters the design can sustain steady state heat loads higher than 10 MW/m². The detailed design and main calculation results are presented in the paper.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 39 / 365

Winding Design for CFETR Central Solenoid Model Coil

Authors: Houxiang HAN^{None}; Yu WU¹; Dapeng Yin¹

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The central solenoid (CS) model coil for China Fusion Engineering Test Reactor (CFETR) is being developed in Hefei, China. The design value of the highest magnetic field for CS model coil is 12T, which is made of Nb₃Sn and NbTi CIC conductor hybrid superconducting magnet. The structural parameter of all windings has been confirmed based on the electromagnetic design and optimization of coil. And the windings for CS model coil has also been designed, 5 windings all are kind of pancake coil composed of layers concentric turns, layer joggles and upper & lower leads.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 59 / 366

The analysis of shielding performance for toroidal field coils of CFETR

Author: Wei Shi^{None}

Co-authors: Qin Zeng ; Wei Li ; Hongli Chen

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Chinese Fusion Engineering Testing Reactor (CFETR) is an ITER-like superconducting TOKAMAK aiming to bridge the gap between ITER and future fusion power plant. Superconducting coils of CFETR provide high-intensity magnetic field to confine the core plasma. Ports are used for RH (Remote Handling) maintenance, plasma diagnose and other measuring equipment. Neutrons leaking from the ports will activate the material and deposit nuclear heat on coils, which may cause coils losing superconductivity. Among superconducting coils, toroidal field coils (TFCs) are closest to the core plasma and suffer the neutronic radiation damage more easily. In order to ensure stability of CFETR operation, shielding analyses of TFCs need to be estimated detailedly.

In this paper, the upper ports and equatorial ports are selected to analyze the influence for TFCs by using the Monte Carlo transport code MCNP with nuclear data library FENDL-2.1. The software McCad which is published by Karlsruhe Institute of Technology is used to convert the CAD model of CFETR into the MCNP input file. The nuclear heat and radiation dose of TFCs are calculated to evaluate the shielding performance. And the influence of the different schemes of ports for TFCs is analyzed, which can obtain an optimized port schemes for the shielding performance. The results can be used as some references for CFETR design.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 85 / 367

Deuterium permeation and retention behavior in a martensitic/ferritic steel

Author: Hao-Dong Liu^{None}

Co-authors: Hai-Shan Zhou ; Xiao-Gang Yuan ; Yu-Ping Xu ; Guang-Nan Luo

Reduced activation martensitic/ferritic steel has been selected as the first wall material of ITER testing blanket modules (TBM). The first wall is subjected to hydrogen isotope permeation by the two mechanisms: one is plasma-driven and the other is gas-driven, which may result in tritium safety and extraction issues. Meanwhile, to evaluate hydrogen isotope permeation and inventory in the first wall material, accurate measurements of hydrogen isotope transport parameters are essential. In present work, deuterium (D) transport parameters including permeability, solubility, diffusivity and recombination coefficient of a martensitic/ferritic steel have been investigated. D retention behavior of the steel exposed to D₂ gas and D plasma has also been compared.

D gas-driven permeation (GDP) through the steel has been performed in a temperature range of 650-800 K to obtain the D transport parameters such as diffusion coefficient, Sieverts' constant and permeability. To evaluate the D recombination coefficient, low energy (several eV) plasma-driven permeation (PDP) has been done in a steady-state laboratory-scale linear plasma device PREFACE at ASIPP. The surface morphology was examined by scanning electron microscopy (SEM) before and after D plasma irradiation. D retention properties of the steel were checked by thermal desorption spectroscopy (TDS) after GDP and PDP experiments.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 69 / 368

Deuterium transport and retention in a liquid metal Gallium under steady state plasma bombardment

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Power and particle handling in the plasma edge region is one of the critical issues, affecting the successful operation of a steady state magnetic fusion power reactor. Tungsten has widely been employed for plasma-facing components in existing fusion experiments and is envisaged to be used for the ITER divertor 1. Unfortunately, conventionally available tungsten is known to suffer from cracking due to its high DBTT. The use of liquid metals has been proposed and implemented in a number of medium-sized confinement devices. Liquid lithium covered divertor was tested on NSTX, and continuously flowing liquid lithium limiter with a loop was performed in EAST, both of which yielded improved plasma performance. Free-falling liquid gallium drops were tested on the T-3M and ISTTOK tokamaks, where no severe effects on the main plasma parameters have been found.

Hydrogen isotopes transport in plasma facing metals is essential to study hydrogen recycling, tritium retention, and tritium recovery. While hydrogen isotopes transport and retention in solid plasma facing materials have been studied in detail, very few information is available for liquid metals. To evaluate these properties for liquid metals, a new technique for the plasma-driven permeation (PDP) experiment has been developed, by holding a liquid metal on a sheet mesh with surface tension. This technique has been verified with a low melting point liquid metal alloy Ga₆₇In_{20.5}Sn_{12.5}, and hydrogen and deuterium diffusivity, surface recombination coefficients have been obtained [2].

In the present work, deuterium PDP experiments for gallium have been conducted in a laboratory-scale linear plasma device, VEHICLE-1. In these PDP experiments, the ion bombardment flux is set of the order of 10^{16} cm⁻²s⁻¹. A liquid metal sample is fixed in such a way that the upstream surface is exposed to deuterium plasma, while the downstream side is pumped to ultrahigh vacuum (10^{-6} - 10^{-5} Pa). The depth of the liquid metal is about 4mm. Deuterium diffusivity and surface recombination coefficients have been obtained by fitting the permeation breakthrough curves. After PDP, deuterium retention in the gallium sample is measured by TDS (Thermal Desorption Spectroscopy). Detailed results will be presented and discussed in the conference.

1 Y. Hirooka et al, Fusion science and technology, Vol.68, 477-483(2015)

[2] H. Bi et al, A study on hydrogen isotopes transport in a liquid metal GaInSn by plasma-driven permeation method, paper presented at 22nd PSI conference, Rome, 2016

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 111 / 369

Hydrogen isotopes plasma-driven permeation through sputter-deposited tungsten coated F82H

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Tungsten (W) has been proposed as the candidate plasma-facing material for the divertor of International Thermonuclear Experimental Reactor (ITER) because of its beneficial properties such as high melting point, high thermal conductivity and low sputtering yield [1]. For a DEMO reactor, surface coatings made of W are necessary to protect the plasma-facing wall made of reduced activation ferritic steels (RAFS) such as F82H (Fe-8Cr-2W) [2]. The characterization of hydrogen isotopes transport through W coated RAFS under plasma exposure is of crucial importance to evaluate major reactor design issues including tritium retention, breeding feasibility and first wall particle recycling. In this study, hydrogen isotopes plasma-driven permeation (PDP) through sputter-deposited W (SP-W) coated F82H has been investigated in the temperature range of 300 - 550 °C using a laboratory-scale linear plasma device: VEHICLE-1 [3]. The plasma density is of the order of 10^{10} cm^{-3} and the net implantation flux is estimated to be $\sim 1 \times 10^{16} \text{ atom/cm}^2/\text{s}$. The incident ion energy is controlled by biasing the sample. A bias of 100 V has been used for PDP in the present work. The density of SP-W coatings is evaluated to be $\sim 19.2 \text{ g/cm}^3$, $\sim 99.5\%$ of bulk W. The thicknesses of W coatings are 0.5 - 4 μm , while the thicknesses of F82H membranes are 0.5 - 2 mm. Bare F82H membranes with various thicknesses are used for comparison.

It has been found that hydrogen isotopes PDP fluxes through SP-W coated F82H are significantly higher than that through bare F82H in the investigated temperature range of 300 - 550 °C. Notice, however, that the PDP flux decreases with increasing temperature. Deuterium retention analyses are performed by thermal desorption spectroscopy (TDS) after steady-state deuterium PDP experiments. The amount of retained deuterium in SP-W coated F82H is a factor of 3 higher than that of bare F82H in the lower temperature range of 320-420 °C and the differences become smaller with increasing temperature. Characterization analyses indicate that enhanced PDP fluxes and high deuterium retention are related to the microstructure of SP-W coatings and the surface recombination characteristics. Nevertheless, further investigations are still underway to address these issues.

[1] R. Pitts, S. Carpentier, F. Escourbiac, et al., J. Nucl. Mater. 438 (2013) S48-S56.

[2] T. Otsuka, T. Tanabe, K. Tokunaga, J. Nucl. Mater. 438 (2013) S1048-S1051.

[3] Y. Hirooka, H. Ohgaki, Y. Ohtsuka, et al., J. Nucl. Mater. 337-339 (2005) 585-589.

Eligible for student paper award?:

Yes

R.OP1: Diagnostics and Instrumentation II / 370

Neutron Diagnostics in the Large Helical Device

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Deuterium plasma experiment begins in the Large Helical Device (LHD) in March, 2017 in order to explore higher-performance plasmas in LHD and to gain a positive prospect toward an LHD-type fusion reactor. In the operation of LHD, neutron yield measurement is essentially required because neutron yield has to be managed in compliance with permitted neutron budgets. In terms of plasma physics, extension of energetic-particle physics can be expected in comparison with the hydrogen regime because neutrons dominantly produced by beam-plasma reactions become newly available

as a signal to diagnose energetic-ion behavior. A comprehensive set of neutron diagnostics has been prepared toward the deuterium operation of LHD. These diagnostics consist of ex-vessel neutron flux monitor (NFM), neutron activation system (NAS), vertical neutron camera (VNC), scintillating-fiber 14 MeV neutron (Sci.-Fi.) detectors, neutron fluctuation detectors, and γ -ray detectors. The NFM on LHD consists of three detector sets. Each set has two different thermal neutron detectors, i.e., a fission chamber (FC) and a high-sensitivity thermal neutron detector. We have newly developed a digital signal processing unit (DSPU) for FC based on the FPGA by using leading edge technologies. The DSPU has both functions of pulse counting and Campbell modes, providing a wide dynamic range up to 9.5×10^9 (cps). In situ calibration of NFM was performed in November 2016 by using a ^{252}Cf neutron source of 800 MBq. To simulate a ring-shaped neutron source, we installed a railway along the magnetic axis position and ran a train loaded with ^{252}Cf . As a result of this work, the calibration factor to evaluate total neutron emission rate (n/s) from pulse counting rate (cps) was obtained with the help of MCNP6 code. The NAS on LHD has two irradiation ends, which perform important roles in cross-checking neutron yield evaluated by the NFM and in investigating shot-integrated 1 MeV triton behavior through measurement of secondary 14 MeV neutron fluxes. The VNC plays an important role in studying radial beam ion transport induced by intrinsic magnetic field ripple and/or external magnetic field perturbation. A neutron collimator is essential for the VNC. Radially aligned eleven cylinders of 3 cm ϕ and 150 cm long embedded in a heavy concrete stack are embedded in the 200 cm concrete floor of the LHD torus hall. Our fast-neutron detector for the VNC is based on the stilbene scintillator and a leading edge fast digitizer equipped with the FPGA. The system was designed so as to realize wide dynamic range capability over 10^6 (cps), having automated n- γ discrimination capability by using the FPGA technology. Sci.-Fi. detectors are installed to study time-resolved 1 MeV triton confinement by measuring secondary 14 MeV neutrons. In addition to these, neutron fluctuation detectors are installed to follow rapid change of neutron flux due to MHD events. Also, CsI(Tl) detectors for γ -rays are prepared for verifying knock on ion-tail formation due to beam-injected fast protons. In this paper, overview of neutron diagnostics prepared for the LHD deuterium experiment, commissioning of these diagnostics, and initial measurement results are described.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 40 / 371

Structural Concept Design Of CFETR CS Model Coil

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The structural concept design of the central solenoid model coil of China Fusion Engineering Test Reactor has been carried out by ASIPP. The CS Model Coil shall produce a 12 Tesla peak field in the bore of the magnet, and the largest magnetic field change rate is 1.5T/S, and operating current is 47.65 KA. CFETR CS Model coil Structural concept design mainly including coil winding design, buffer zone concept design, preload structure concept design, coil joint and joint support concept design, helium supply system and pipe support concept design and so on.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 29 / 372

Swirl tube design of the pole shiled in the magnet for the long pulse upgrades of EAST-NBI based on the subcooled boiling

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Neutral Beam Injection (NBI) auxiliary heating system for Experimental Advanced Superconducting Tokamak (EAST) designed with the design of the 180 degree magnetic field deflection to deflect the un-neutralized particles during beam transmission. In order to protect against the divergent beam, the copper pole shields are placed on both sides of the neutral beam in front of each deflection magnet. For future planned EAST-NBI operation with longer beam pulse lengths (more than 100 s to 1000 s) and higher beam power (2~4 MW), heat density will be more than 10 MW/m² deposited on each shield, so it is very important to enhance the heat transfer of shields to ensure their good working performance and even the safety operation of NBI system. In this paper, based on the analysis of the heat transfer performance of the initial structure of the EAST-NBI deflection magnet pole shield, the heat transfer enhancement design with swirl tape inserts of cooling tube is proposed, called swirl tube. Considering the heat flux distribution mechanism of Rensselaer Polytechnic Institute (RPI) model, interfacial mass transfer, interfacial momentum and energy transfer and the empirical relation of void fraction, combined with Eulerian two-phase flow and heat transfer control equations, the calculation method and flow chart of subcooling boiling was improved. Then the thermo-flow-solid coupling analysis has been done to the pole shield structure with swirl tubes, and the performance optimization of its key structural parameters was completed according to the operation limitation of the pole shield and its cooling water system. The heat transfer performance of the final structure was analyzed and checked well meeting the design requirements. This study is a theoretical basis of the experimental design of heat transfer enhancement structure for the pole shield in EAST-NBI system, and provides a reference for the heat transfer enhancement design of other high-heat-flux components in the EAST-NBI system, so it has a very important theoretical and practical significance for future long pulse and high power operation.

Keywords—Neutral beam injection, High heat flux components, Swirl tube, Heat transfer enhancement, Thermo-flow-solid coupling analysis

ACKNOWLEDGMENT

This work is supported by the Foundation of ASIPP (Contract No. DSJJ-15-GC02) and the National Natural Science Foundation of China (NNSFC) (Contract No. 11605234)

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 4 / 373

Coupling analysis of the HCCB blanket under electromagnetic, thermal and mechanical loads

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The China Fusion Engineering Test Reactor (CFETR) is a superconducting tokamak proposed by the China National Integration Design Group. The missions of CFETR are achieving 50~200MW

fusion power, and steady-state operation with the duty cycle between 0.3 and 0.5. Breeding blanket, a core component of fusion reactor, takes the role of tritium breeding and energy conversion. It is located inside the vacuum vessel under a complex operation environment. During the steady-state operation, the blanket is not only affected by its gravity but also a large amount of nuclear heat from the plasma, especially under the plasma major disruption or plasma vertical displacement. Variation in magnetic flux will produce a huge eddy current and electromagnetic force in the blanket and seriously affects the structural safety of the blanket. A comprehensive finite element (FEM) structure analysis has been performed to evaluate the rationality of the blanket structure design. The eddy current and Lorentz forces caused by plasma major disruption from an initial current of 10 MA to zero in 36 ms have been calculated. The mechanical loads due to gravity have also been accounted and discussed. Meanwhile, the thermal analysis of the blanket was carried out by means of the nuclear thermal data given by MCNP. All the above loads have been combined as input for a FEM analysis and the stress distribution has been evaluated according to the American Society of Mechanical Engineers (ASME) norms.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 13 / 374

Evaluation of spatial resolution of neutron profile monitor in LHD

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Deuterium plasma experiments in the Large Helical Device (LHD) will begin in March 2017. In LHD, neutrons are mainly generated by interaction between bulk plasmas and beam ions. Therefore, neutron emission profile measurement plays an important role in the understanding of confined beam-ion behavior.

The vertical neutron camera (VNC) has been developed to measure neutron emission profile in LHD. The VNC consists of a multichannel collimator made of heavy concrete embedded in the 2 m-thick concrete floor of the LHD torus hall, fast-neutron stilbene-scintillation-detectors, and the digital-based data acquisition system with high-speed sampling rate of 1 GHz and online/offline neutron-gamma discrimination capability.

The spatial resolution evaluation experiment of VNC was carried out in November, 2016 by using a ²⁵²Cf neutron source of 800 MBq. The ²⁵²Cf neutron source was introduced into the vacuum vessel through an aluminum pipe from an upper diagnostics port. There were two source positions. One is just on the collimator axis at major radius R=3,450 mm (case A), and the other is in the middle of two neighboring collimator axes at R=3,405 mm (case B). The positions of stilbene scintillation detectors #1, #2, and #3 were located in R=3,360 mm, 3,450 mm, and 4,260 mm, respectively.

In case A, measured neutron count rates of detectors #1 and #2 were 0.0219 cps and 2.45 cps, respectively. Neutron transport calculations using a general-purpose Monte Carlo Neutral Particle code 6 (MCNP6) indicated that the neutron count rate of detectors #1 and #2 were 0.060 cps and 2.3 cps, respectively. Here, efficiency of stilbene detectors for ²⁵²Cf neutron has been evaluated to be 0.20 counts/(neutron/cm²) with experiment. In the MCNP6 calculation, the integrated value of flux is calculated in the range of 700 keV-16 MeV, which is evaluated from experimental results in the Fast

Neutron Laboratory at Tohoku University. In case B, measured neutron count rates of detectors #1 and #2 were 0.498 cps and 0.542 cps, respectively. In the MCNP6 calculation, the neutron count rates of detectors #1 and #2 were 0.68 cps and 0.61cps, respectively. Thus the experimental results agree well with the MCNP6 calculation. We confirm that VNC has sufficient spatial resolution for study of fast-ion radial transport.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 5 / 375

Corrosion test results of ARAA and FMS steel in the Experimental loop for liquid breeder

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A helium cooled liquid lithium (or lead lithium) concept has been developed to design a liquid breeder blanket in Korea. Ferritic-martensitic steel (FMS) was selected as a structural material for fusion reactors, and a commercial-scale Advanced Reduced Activation Alloy (ARAA) has been developed. An Experimental Loop for a Liquid breeder (ELLI), which we designed and fabricated ourselves, was constructed to validate the electromagnetic (EM) pump as well as to test the effects of the magneto-hydrodynamics (MHD) of a liquid metal flow, and to investigate the compatibility of PbLi with the structural materials of FMS. Thus far, performance tests on each component, such as the heaters and control systems used for heating the loop, were conducted, and characteristic tests using a magnet and an EM pump were carried out. Corrosion tests using ELLI were conducted using grade 91 FMS steel and ARAA for a 250 hour period (100+150 hours). In this study, a commercial FMS with grade 91 and developed ARAA were used for corrosion test-specimens to compare the corrosion characteristics in the flowing PbLi loop. A corrosion test was conducted for investigating the compatibility of PbLi using structural materials. The effects of the oxide coatings on the prevention of the corrosion progress were also investigated.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 74 / 376

Signal transmission links for the electron cyclotron resonance heating system on J-TEXT

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Electron Cyclotron Resonance Heating (ECRH) system is an important auxiliary heating method which is widely used in magnetic confinement fusion. For the ECRH system on J-TEXT, signals should be transmitted to the control system for monitoring and protection, and also to the data acquisition system. Considering the high voltage and harsh Electro Magnetic Interference (EMI) environment, reliable fiber optic transmission links are applied. Transmitted by fiber-optic modules, the response time of digital signals can be less than 1 μ s. According to the ECRH system requirements, two practical analog fiber optic transmission links have been designed and developed. Using AD650, the delay time of most analog signals is less than 100 μ s. Based on the Voltage-to-Frequency (V/F) and Frequency-to-Voltage (F/V) conversion technology with VFC110, critical signal transmission delay can be shortened to 5 μ s. The test results indicate that the designed signal transmission links for ECRH system on J-TEXT are stable and reliable.

Keywords: ECRH, Fiber optical link, Signal conditioning

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 106 / 377

The Analysis of Socio-economic Impact on Big Science R&D: Focusing on Fusion R&D Program in Korea

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This paper is focused on the analysis of socio-economic benefits of the ongoing R&D program on big science such as nuclear fusion in Korea. The spillover effects are understood here as positive externalities of publicly funded R&D activities that may be revealed at the companies' level in the form of newly created knowledge stock; development of innovative products/ processes with broader market applications; strengthening of R&D, manufacturing and marketing capabilities; etc. In addition, this study critically reviews the literature on the socio-economic benefits of publicly funded big science R&D. In that literature, two main methodological approaches have been adopted —surveys and case studies. These studies have also highlighted the importance of spillovers and the existence of localization effects in research. From the literature based on surveys and on case studies, it is clear that the benefits from public investment in big science R&D can take a variety of forms. We classify these into seven main categories, reviewing the evidence on the nature and extent of each type. The results demonstrate that fusion R&D programs have relatively outstanding performance in seven categories: (1) increasing the stock of useful knowledge; (2) training skilled graduates and researchers; (3) creating new scientific means and methodologies; (4) forming networks and stimulating social interactions; (5) reinforcing the capacity for scientific and technological problem-solving; (6) creating new firms; and (7) access to scientific facilities. In particular, those projects were observed to form an industrial ecosystem for nuclear fusion that extends to the accelerator sector, in the category of creating new firms, while making a significant contribution to training talented researchers and expanding social networks as well. We reconsider the rationale for government funding of big science R&D, arguing that the traditional 'market failure' justification needs to be extended to take account of these different forms of benefit from big science R&D. The article concludes by identifying some of the policy implications that follow from this review.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 41 / 379

Research and Analysis on Electrical Performance of EAST Cryogenic Axial Insulation Breaks

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Abstract: The cryogenic axial insulation breaks (IBs) are key components of the superconducting magnet system for the EAST device, which play an important role of the liquid nitrogen and liquid helium insulation channel. In order to ensure the safe operation of the IBs during the working life of the EAST device, experimental research and analysis on electrical performance should be carried out, including structural analysis, partial discharge test, direct current high voltage test in helium pressure and impulse test.

Keywords: Axial insulation breaks; Structural analysis; Partial discharge; Impulse.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 53 / 380

Micro perspective on anti-fatigue performance enhancement of PFC metal welding interface with MD simulation

Authors: Shenghong Huang^{None}; Xuan Wang^{None}

As one of key technical specification, the anti-fatigue performance of plasma-facing components (PFC) in fusion reactor receives widely concerns. Many researchs concluded that the micro structure on PFC metal connecting interface greatly affects its fatigue performance of interface, especially for some micro/nano scale pores in connection zone, which are difficult to be detected by ordinary techniques and easy to be neglected. In this paper, a method of utilizing impacting stress wave to eliminate internal nano pores on welding interface of PFC component is proposed. To examine the feasibility and results of this method, molecular dynamics models are established. Then the healing process of a half sphere nano cavity under the impacting of different stress wave at different temperature conditions are computed and observed. To remedy the dislocation of grain after impacting stress wave, a kind of post heat treatment process is simulated based on different nano cavity-healed results at different conditions. Finally, the fatigue strengths of different cases are compared. The results show a prominent increase of fatigue strength for case treated by proposed method.

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 54 / 382

Application of automatic ultrasonic testing system based on joint robot in Fusion Engineering

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The articulated arm robot has precise mechanical link, similar to the human arm and it is integrated with the ultrasonic testing system, which can provide users with a flexible automatic ultrasonic testing scheme. The Institute of Plasma Physics, Chinese Academy of Sciences, has introduced a multi joint mechanical arm ultrasonic testing system from the French M2M company, and developed a variety of detection methods based on the system. This paper focuses on the application of the technology in ITER Feeder explosive welding composite plate, EAST W/Cu divertor tube plate weld, CFETR vacuum chamber austenitic stainless steel welding seam detection. The results of test show that the automatic ultrasonic inspection system which based on joint robot has the characteristics of flexible system, high detection efficiency and good repeatability and it will have a wide application prospect in fusion engineering.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 14 / 383

A preliminary consideration of CFETR diagnostic system

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Chinese Fusion Engineering Test Reactor (CFETR), which is under conceptual design to bridge gaps between ITER and DEMO, is envisioned to produce a fusion power (50-200 MW for phase I and up to 1GW for phase II) with tritium breeding ratio (TBR) \geq 1.0 and a duty cycle time of approximately 0.3-0.5. This presentation will introduce the current work for the conceptual design of CFETR diagnostic system. Based on the experience obtained in the development of ITER diagnostics and combined with CFETR machine requirement, some preliminary considerations for CFETR diagnostic system have been described. They mainly includes: 1. the conditions and constraints in CFETR environment. 2. First considerations for the conceptual design. 3. One list of proposed measurements and candidate diagnostic techniques for CFETR early phase with discussions on the possible R&D activities. At the end, several issues have been discussed and the plan of future work will be outlined.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 76 / 385

Design and optimization of cooling channels for 4-strap ICRF antenna of EAST

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In order to achieve high requirement of EAST Tokamak, ion cyclotron range of frequency(ICRF) heating is utilized as one of the main auxiliary heating methods, which plays an important role in the coupling RF power to the plasma. During the operation of ICRF, there will be a large amount of heat flux on the surface of antenna and consequently the structural stability and reliability of the antenna become worse. Therefore, the cooling channels of the ICRF antenna is designed based on the RF loss on the antenna, and optimization of the cooling channels is also finished by finite element method.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 42 / 386

Design and Analysis of Magnet System for Flii Testbed in EAST

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Abstract: To simulate the magnetic field environment in EAST Tokamak, and to study the flowing of liquid metal driven by electromagnetic force, and then to guide the liquid lithium limiter experiment in EAST preferably, it's necessary to develop a magnet system. In order to ensure the experiments are carried out smoothly, the running time of magnet system should be more than 1000s, the background magnetic field should be adjustable within the scope of 0~2T, and the uniform magnetic field must be greater than 50mm×50mm. In this paper, the design and checking calculation of Helmholtz coil and electromagnet was carried out by ANSYS and MAXWELL respectively, to ensure the safe and stabilized operation of magnet coil, and to minimize the cost the testbed. Base on the analysis results, the yoke, ampere turns and radius of the magnet coil have significant effects on the magnetic field distribution, the parameters of power and cooling system, and the comprehensive cost of the testbed. Considering electrical and hydraulic parameter, and total cost of magnet system, A C-frame electromagnet was adopted whose number of turns are 16×8, and radius is 0.56m.

Keywords: EAST; Flii Testbed; Magnet System; ANSYS

Eligible for student paper award?:

Yes

T.POS: Poster Session T - Board: 55 / 387

Repair of the cracked surface of W using high energy pulsed laser

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W is a promising plasma facing material for fusion devices. It is expected to suffer from the transient heat flux during normal ELMs, abnormal VDEs and plasma disruption events. In the next fusion device such as the ITER, the transient high heat flux can reach up to several MJ/m² in a very short pulse (~ms), which is as high as enough to cause the surface damages especially in form of cracks [1]. The cracks, to some extent degrade material performance. To alleviate the influence of the cracks, the repair of the cracked surface using high energy pulsed laser has been proposed and investigated [2].

In the present work, the repair of cracked surface of W was performed in the laser and wall material evaluation device. The W was pre-damaged by the transient heat flux exposure in EMS-60 with parameters of 400MW/m² for 1ms and 100 cycles. The net-like macro cracks were successfully generated as expected. Then, the damaged surface of W was repaired in the laser and wall material evaluation device with a high vacuum circumstance. The damaged W was preheated to a elevated temperature exceeding DBTT of W, then the high energy pulsed laser with a wavelength of 532 nm, a energy of 0.2~0.8 kJ, a frequency of 10 Hz and a circular beam diameter about 0.4 mm was scanned on the damaged area with the adjustable single spot repetition numbers and overlapping ratio between adjacent spots from 10-50%.

After the repair process, the net-like cracks successfully disappeared at the laser scanned areas, meanwhile, there were no any other type cracks founded. It is should note the high energy pulsed laser can also cause other type of crack patterns at room temperature. Thus, the pre-heating process suppressed the cracks formation by laser shocks. The residual trace for the net-like cracks could be also distinguished by micro observation. In addition, the surface seemed to become rough from micro perspective, identifying that the surface underwent the plastic deformation during laser scanning. The repair mechanism may different with the laser re-melting method with an initial room temperature [3]. The single spot repetition numbers, the overlapping ratio between adjacent spots, the laser energy and spot dimension have the important influence on the repair effect and need in-depth optimization. Moreover, the behavior and properties of the repair surface under the subsequent plasma and heat exposure is unknown and need future relevant tests.

[1] Xiang Liu, Youyun Lian, Lei Chen, et. al., Experimental and numerical simulations of ELM-like transient damage behaviors to different grade tungsten and tungsten alloys, *J. Nucl. Mater.* 463 (2015) 166-169.

[2] Y. Ueda, Pulsed heat load effects on plasma facing materials, 2015 ITER International School, Hefei, China, 2015,14 - 18

[3] Th. Loewenhoff, J. Linke, J. Matějček, et. al., Laser re-melting of tungsten damaged by transient heat loads, *Nuclear Materials and Energy* 9 (2016) 165–170

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 15 / 388

The management and storage of EAST diagnostic data

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For the purpose of plasma physics research on tokamak experimental, many diagnostics have been developed for Experimental Advanced Superconducting Tokamak (EAST). A distributed and continuous data acquisition system has been implemented for the diagnostic system. At present, there are more than 60 data acquisition units and more than 2500 raw signals including scientific data, video data, and images. The total maximum data streaming throughput is more than 5GBytes/s. The acquired data are combined into segment data in several seconds and transferred into data servers, and all the data are continuously archived into EAST database basing on mdsplus during discharge. The EAST mdsplus database are organized as several distributed trees, and each tree is composed into several sub-trees depending on different diagnostics.

The whole data storage system is constructed into 3 layers of different performance and capacity, latest data, archived data, and backup data. The layer-1 is the latest data stored into several servers

installed with PCIe SSD which can provide fast IOPS and high read write speed. The layer-2 is a local SAS storage with all the archived data, the data size of last year is about 100TBytes and the total data is about 500TBytes. The layer-3 is a NFS backup data storage located in another data center in order to keep data safety. Researchers can access the diagnostic data using different ways such as browsers and C/MATLAB/Python clients interfaces. A computing server cluster with data access tools has also been provided for data analysis. The system details will be presented in the paper.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 56 / 389

Analysis on Phase array ultrasonic signals of the ITER PF jacket inspection

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ITER magnet system consists of 18 Toroidal Field (TF) coils, a Central Solenoid (CS), 6 Poloidal Field (PF) coils 9 pairs of Correction Coils (CCs). These four types of the coils are relied on Cable-In-Conduit Conductors (CICCs). The CICC for the PF Coils of ITER is referred to as PF conductors. This paper is mainly presented the Phase array ultrasonic test (PAUT) results on PF conductor jackets and analyzed the typical signals based on the statistics. Wavelet Transform (WT) method are proposed to extract signal waveform feature which will be provide reference basis for characterization of defects.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 16 / 390

Use Spectrum Simulation Code SOS to test the performance of the Fast ion D-alpha spectrum on HL-2A

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Use Spectrum Simulation Code SOS to test the performance of the Fast ion D-alpha spectrum on HL-2A

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ABSTRACT: In magnetic confined fusion devices, the fast ion is usually generated in heating process, especially when the neutral beam heating is on. When the fast ions collide with the neutral beam, some fast ions neutralize and emit light, but the intensity levels is usually below the continuum radiation level and several orders of magnitude below the thermal charge-exchange(CX), beam emission spectrum and motional stark spectrum level. In order to investigate the fast ion behavior, a fast ion D-alpha(FIDA)

system has been built on HL-2A. It can only acquire signal in the red shift direction, because there is no other windows suit for the system in HL-2A. It uses the Gauss curves and least mean square method to fit several components consist of different energy (full energy, half energy and third energy) which also has its stark splits. It gets the FIDA signal performance and its simulation results, which consistent with each other.

KEYWORDS: fast ion, charge exchange recombination spectrum, Fast ion $D\alpha$, Beam emission spectrum, Motional stark emission

ACKNOWLEDGEMENT: The author give full appreciate to Professor Manfred von Hellermann for the discussion. South Western Institute of Physics(SWIP) for helpfull advices.wujing@uestc.edu.cn

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 98 / 391

Design & Development of High Voltage Power Supply for Negative Ion Source

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Neutral beam injection(NBI) is an essential plasma heating tool for the China Fusion Engineering Test Reactor(CFETR),that is under engineering conceptual design.The CFETR NBI of energy higher than 500keV is needed. Because of neutralization efficiency of negative ions is higher than that of positive ions under the high energy,the negative ion source is required for NBI,which ask for higher voltage than positive ion.High voltage power supplies(HVPS) is very important power supply for Neutral beam injection (NBI) of fusion experimental device.To simplify the system structure and improve the high accuracy of out voltage, the quantity of the switch power supply(SPS) modules should be as little as possible.Therefore, a HVPS of moderate voltage and number be needed.Then an HVPS for the negative ions NBI is proposed,which must be designed by having a few different voltage classes of SPS in series by link.Meanwhile, the control of SPS of negative ions NBI becomes more complicated and difficult because of the power supply has many different voltage rating.

A set of HVPS with PSM topology at 16kV /20 A has been designed and successfully tested at Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP).The power supply has the characteristics of high stability, fast dynamics, short protective time and low stored energy. This power supply consists of 19 modules of 800V and 8 modules of 100V. 27 series-connected SPS modules are fed from multisecondary transformers.Insulated-gate bipolar transistor (IGBT) is used as the output switch to improve the HVPS dynamic performance.HVPS output voltage is adjusted by the control system.To ensure normal HVPS operation and fulfill the requirements of ion sources, the control system has characteristics of setting voltage preset value and different rise or fall times of voltage of HVPS,blocking the voltage output of the HVPS in case of faults,etc.To reduce voltage overshoot by the simulation,a proportional (P) controller is selected to control the out voltage of HVPS. The feedback control system runs on digital signal processor (DSP) and field programmable gate array(FPGA).To isolate the high potential and avoid the electromagnetic interference,all the control circuit interfaces are through fiber optic cables for HV isolation.

Dummy load made of resistance is necessary to observe power system performance. The rise time of out voltage can be set from to .Solving the balance between the rise voltage overshoot and rise time,the method is that SPS modules be sequentially turned-on according to the 90% of the value of preset voltage at the setting rise time,and then later other modules be opened step by step at interval until reaching the preset value of output voltage via close-loop feedback control.The test results of dummy load that HVPS complies with the requirements of negative ion NBI Extraction.It is planned to assemble the negative ion source and the HVPS, then start the experiment of negative ion extraction in the next stage.This power supply will be extended at more voltage ratings or higher voltage ratings and applied to an accelerator system for negative ion sources NBI HVPS in the future.

Eligible for student paper award?:

Yes

W.OA1: Materials II / 392**Synergetic effects of He ions irradiation and oxidation on W**

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Tungsten (W) is foreseen as one of the most promising plasma facing materials (PFM) of fusion devices. As oxygen exists as a contamination in the vacuum chamber of tokamak, W oxide can form at the surface of W due to good chemical affinity between W and oxygen. W oxide film will inevitably interact with helium (He) ions, which are formed by the deuterium–tritium (D-T) fusion reaction. Many studies have found that He ions irradiation causes degradation of W physical and mechanical properties, such as the thermal conductivity (TC), which is related to the surface modification of W. Surface morphology of W will be changed by the oxidation too. Thus He ions irradiation and oxidation might synergistically degrade the W physical and mechanical properties. However, the synergetic effects of He ions irradiation and oxidation on W are not understood. In this study, thin WO₃ film was produced by thermal oxidation on the surface of pure W samples. The samples were sequentially irradiated by the He ions. The morphology of W after He ions irradiation and oxidation were observed. Finally, TC of He ions implanted layer of W with an oxide film was measured.

In the oxidation process, WO₃ film was produced on the surface of rolled W by thermal oxidation at 673 K and at an oxygen pressure of 590 torr for 90 min. X-ray diffraction confirmed the main phase of the oxide was WO₃. A damaged layer in the near surface region of W with an oxide film was produced by He ions at room temperature. The dose were simulated by SRIM. Three dose were implanted and damaged layer of average 0.1, 0.5 and 1 dpa were obtained. The morphology of the samples before and after irradiation was observed by scanning electron microscope. The TC of the implanted layer was measured by front heating transient thermosreflectance method. The surface morphology and TC of the He ions implanted layer were compared and analyzed.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 70 / 393**Performance of full compositional W/Cu functionally gradient materials under quasi-steady state heat load**

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Tungsten (W) is the most promising plasma facing material for fusion devices, while copper (Cu) has been proposed as the heat sink material behind plasma facing material¹. Nevertheless, because of the large difference of coefficient of thermal expansion (CTE) between W and Cu, the joining of these two dissimilar materials causes the high thermal stress concentration at the interface when exposed to the high heat loads. The stress concentration leads to the interface cracking, reducing the lifetime of the components. To mitigate such damage, the W/Cu functionally graded layers between W and Cu have been proposed to provide a smooth transition of properties and thus alleviate the thermal mismatch [2].

To investigate the performance of graded structure materials under fusion relevant steady-state heat load, six-layered W/Cu functionally gradient materials (FGMs) with full compositional distributions (from 0 to 100%) prepared by resistance sintering under ultra-high pressure (RSUHP) method were tested under the electron beam material testing scenario (EMS-60) using the fusion relevant heat load in the range of 6.5~9.8 MW/m². The specimens were loaded for 5s, and then cooled for 30 s for indirect cooling. Meanwhile, the W/Cu mock-ups without any transition layers were also fabricated and tested for compare. In addition, the finite element simulation of the thermal-stress evolution and distribution was performed for analysis.

After 50 cycles 6.5 MW/m² and 10 cycles 8.2 MW/m², there are no obvious changes of the heat transition for both W/Cu FGM and W/Cu mock-up. Only a local small crack at graded interface of W/Cu FGM was founded by microscopic observation, however, the visible crack through along the sharp interface was observed. And, the residual stress distribution along the edge side for W/Cu FGM is about lower 50% than that for W/Cu mock-up. Both the experiment and simulation identified that the graded transition layers can ameliorate the stress concentration and alleviate the thermal mismatch. However, after the high heat load of 9.8 MW/m², both the W/Cu FGM and W/Cu mock-up show the obvious deterioration of the heat transition in a few cycles, in which the W/Cu FGM shows the severe melting and infiltration of the Cu and also the cracking along graded interface, while the W/Cu mock up shows the severe crack along interface and partly exfoliation. These reveal that it is necessary to design a thicker W layer for the high temperature gradient to protect the behind gradient layers, and the W layer processed by RSUHP method needs to be improved by optimizing the fabrication parameters.

1 Jeong-Yong Park, Yang-II Jung, Byung-Kwon Choi, et al., *J. Nucl. Mater.*,417 (2011) 916 - 919

[2] G. Pintsuk, S. E. Brünings, J. E. Döring, et al., *Fus. Eng. Des.*, 66 - 68 (2003) 237 - 240

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 86 / 394

DEVELOPMENT AND VERIFICATION OF COMPUTATIONAL MODEL FOR CONTROL OF PLASMA AND HALO CURRENT IN EAST TOKAMAK

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Co-authors: Salah Ud-Din Khan ; Song Yuntao ; Muhammad Shoaib ; Ghulam Farid

A new technology for developing fusion energy is to use hydrogen isotopes i.e., deuterium (D) and tritium (T). It is a combine effort for building up of International Thermonuclear Reactor (ITER) named as Tokamak, which will come into operation in 2020. Handsome amount of work has already been done by many researchers contingent with plasma shape, halo current and plasma equilibrium properties by numerical techniques. Halo current calculation and plasma stability is the most important problem in fusion technology. It seems to be an open boundary problem in which a system is designed in order to compel the plasma in it designated orbit. There encounter large degrees of freedom in shaping hot plasma in torus form as long as symmetry in magnetic field is maintained in specific magnetic flux surfaces. Therefore, non-linear plasma coupled system model have been developed which allows an effective way to study the chaotic behavior of the main plasma characteristics under controlled conditions. Keeping this criteria, theoretical approach has been developed

in order to evaluate the new innovative model for calculating different aspects and stability of Tokamak reactor. Stability achieved in Tokamak includes creating balance between pressure and forces due to magnetic field and to setup the shape and position of plasma. The developed system compels the plasma in same orbit and associated two isolated equilibrium points of Tokamak for gaining the stability and instability position via theoretical approach. It can be much convenient to calculate the chaotic behavior of Tokamak. These models also provide the total magnetic field, asymmetric forces, conducting points and poloidal halo current. Finally, experimental data will be comparing by taking the difference between the two theoretical models. For the critical analysis of all aspects of this research, a non-linear system has been developed and is given as,

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 43 / 395

Qualification of ITER Correction Coil Superconductor Joint

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The system of Correction Coils (CC) is a component of the ITER Magnet system, required to correct toroidal asymmetries and reduce error magnetic fields detrimental for physical processes in the plasma. Coil terminals will be connected to feeder terminals using twin-box joints. Qualification of the manufacturing procedure of the coil terminals is achieved by performing DC tests of prototype joints in relevant conditions of current, temperature and background field (4.5 K, 10 kA, 2.5 T).

In order to control the DC resistance and AC loss, special tooling and processes were developed. The main processes of CC joint are Conductor de-jacket, Nickel removal, Silver and TIG welding. Two samples were manufactured in 2015 and 2016 for measuring the DC resistance and AC loss. Joint sample performance was analysed after 1000 electromagnetic cyclic loading. The tests were carried out by SULTAN facility and ASIPP facility. All the measurement results shows CC joint samples DC resistance under 5nΩ, AC loss under 7J/cycle.

Index Terms —ITER, Correction Coil (CC), joint, qualification.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 6 / 396

Evaluation of tritium inventory and permeation in water-cooled ceramic breeder blanket for CFETR

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Demonstration of tritium self-sustainability is among the key targets of the China Fusion Engineering Test Reactor (CFETR). The water-cooled ceramic breeder blanket (WCCB) is one of the blanket candidates for CFETR. However, tritium retention and permeation in blanket systems can become a bottleneck for the design and operation of future fusion devices with respect to economic and safety concerns. Hence, first attempts have been made to evaluate tritium inventory and permeation in the WCCB blanket for CFETR by two parallel approaches—the conventional lower dimensional diffusion simulation reported in this paper, and the multi-dimensional calculation based on the Finite Element Method (FEM) reported in a separate paper. In the first approach, a 1D model comprising the first wall and the breeding region is developed for the WCCB blanket, with simplifications to reduce complexity yet maintain desired accuracy. Tritium permeation and retention calculations are carried out using the TMAP code. Both the pulse and steady-state operation modes are simulated. The present simulation takes into account the effect of surface conditions, temperature gradient, and trapping in defects. The results show that the amount of tritium permeated is significantly reduced by using a tungsten armor at the front side of the first wall. Simulation results also indicate that in order to meet the designation criteria for tritium permeation, it is necessary to apply coating technology to blanket coolant channels.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 77 / 397

Development of off-axis beamline for KSTAR

Authors: Sangwook Jung¹; Sonjong Wang¹; Jinhyun Jeong¹; ByungKeun Na¹; Seung Ho Jeong²; Tae seong Kim³; Jong-Gu Kwak⁴; Young kook Oh¹; Yaung soo Kim¹

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Present 6 MW NBI system is one of the important heating device in KSTAR. In 2016 campaign, the system contributed to high betaN, and almost steady-state operation of 78.3 seconds. In addition to the present system, the KSTAR has plans to enhance heating and current drive. For this plan, one more 6 MW NBI system will be installed in 2017, and will be operated from 2018. Two of three ion sources are for an off-axis current drive, and the other one is for an on-axis current drive. The target operating pulse duration is 300 seconds. The neutral power of each ion source is 2MW. The vertical slant angle of the off-axis ion source is 5.5 degree, and the beam tangency radius is 1.56 m from the axis of the tokamak. The beam size at the ion source exit grid is 450mm×130mm. The beam comprises 280 beamlets, and the partial beamlets are focused at 10 and 12 m vertically and horizontally, respectively, from the ion source in order to increase the beam transmission efficiency. A beam transport characteristic has been predicted with beam transport code. The code can simulate neutral beam transport, loss, ionization, and heat load of beamline components. The heat load was calculated assuming that the fractional energy current fraction of the neutral beam is 8: 1: 1 (full energy: half energy: third energy). The transmitted beam powers were 2.14 MW of full energy, 0.134MW of half energy and 0.095MW of third energy. Total injected neutral beam power would be 2.284MW, and total transmission efficiency was estimated to be 84.1%. The peak loaded beam density on the whole beam line components except a residual ion dump and a calorimeter was below 4MW/m². The highest residual ion beam load onto a full energy dump was expected to be 6.7 MW/m². The designed calorimeter is a hinged door structure. When the calorimeter was closed, the peak loaded beam power density was 6.49MW/m². These results satisfy critical heat flux of hypervapotron, > 10MW/m². Based on these results, the beamline components of new NBI system is being manufactured.

Eligible for student paper award?:

No

T.OA3: Blankets and Tritium Breeding: Liquid Breeders / 398

DESIGN OF CHINESE DEMO BLANKET CONCEPTS AND R&D PROGRESS OF DFLL TBM

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China has long been active in pushing forward the fusion energy development to the demonstration of electricity generation. As one of the most challenging components in DEMO, great efforts have been put on the development of breeder blanket and three blanket schemes were studied in China for fusion engineering test complementary with ITER. In this paper, the main blanket concepts developed in China will be summarized including two leading schemes of Dual Functional Lead Lithium (DFLL) and Helium Cooled Ceramic Breeder (HCCB).

For ITER-TBM, the Procurement Agreements of HCCB-TBM has now been confirmed and led by three institutes, i.e. SWIP (Southwest institute of Physics), INEST (Institute of Nuclear Energy Safety Technology) and CAEP (Chinese Academy of Engineering Physics), each playing a different role. The other candidate scheme, DFLL-TBM has also been continuously supported by CN-MOST (Ministry of Science and Technology) and will be tested in ITER under international cooperation.

And the technical challenges to ITER-TBM and also the DEMO mainly focus on fusion material development and testing, breeder/coolant technology and experiment validation, effective tritium production/extraction to achieve self-sufficiency, reliability and safety etc., which are the nuclear technology basis of DEMO blanket. In this paper, the recent R&D progress on DFLL-TBM is presented, including the progress on structural material fabrication technologies and properties of CLAM steel, PbLi/He coolant technology and safety issues, the RAMI analysis, small mockup neutronics experiments, and tritium behavior etc.. Based on the conceptual design and latest technical R&D progress, the entire test planning will be scheduled for DFLL concepts towards DEMO blanket.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 48 / 399

Hydrogen Effects on Properties of ICP Sprayed Boron Carbide Coatings

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Boron carbide (B₄C) is low-Z material with good chemical stability and effective neutron absorption, so it has received attention for application in nuclear fusion reactors and plasma facing material in fusion devices. B₄C coatings are successfully deposited by inductively coupled plasma (ICP) torch, and the results indicate that plasma gas composition has great affection on melting process of B₄C powders and also the properties of B₄C coatings. In this work, the boron carbide coating with low porosity, high binding strength and good performance of thermal shock behavior is deposited by the control of content of hydrogen in plasma gas. Surface morphologies of coatings are characterized by scanning electron microscope (SEM). The porosity is measured by picture of polished coating taken by optical microscope. The binding strength of coating is obtained through tensile test. A system including electron beam welding machine and water-cooling platform is used to test the thermal shock behavior of coating. The results show that adding hydrogen to the plasma gas is able to improve the properties of spray coating. The relationship between the properties of boron carbide coating and content of hydrogen in plasma gas is discussed.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 99 / 400

The SF₆ Gas Handling and Storage Plant of the MITICA test facility

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The ITER Neutral Beam Injector (NBI) is designed to deliver 16.5 MW of additional heating power to the plasma, accelerating Deuterium or Hydrogen negative ions up to -1 MV with a current as high as 46A (for H₂). To prove the feasibility of the NBI system and demonstrate the achievement of the very demanding performance, a dedicated test facility is under construction in Padova, Italy, named PRIMA (Padua Research on ITER Megavolt Accelerator). PRIMA will host a full-scale prototype of the injector, the so called MITICA (Megavolt ITER Injector Concept Advanced) experiment. The main power supply of the MITICA injector is the Acceleration Grid Power Supply (AGPS), which provides the power to the acceleration grids of the injector. The AGPS is a special power supply with high rated power (about 55 MW), extremely high dc output voltage (-1MV dc) and long duration pulses (up to 1 hour). The AGPS output stage is composed by high voltage equipment such as step-up insulation transformers, diode rectifiers and filters. The diode rectifiers are connected in series at the output side in order to increase the dc voltage up to the required value (-1MV for Deuterium, -870kV for Hydrogen). A -1MV dc transmission line, about 100m long, connects the diode rectifiers to the accelerator via a high voltage bushing. Auxiliary services, such as the cooling water and injector gas, are provided to the injector through the Transmission Line itself.

Due to the high voltage level (1MV dc), the high voltage equipment is designed for pressurized SF₆ gas insulation. Diode rectifiers, filters and the transmission line are enclosed within stainless steel tanks to be filled with SF₆ gas at 0.6 MPa. Due to the huge size of the installation, and the consequent required amount of gas (about 30 tons), the high voltage equipment is divided into nine separate SF₆ gas compartments. The management, handling and storage of the SF₆ gas requires a properly designed Gas Handling and Storage Plant (GHSP) which must be able to fill, recover and store the gas at the required pressure from the compartments within a specified time.

This paper describes the design, installation and commissioning of the MITICA SF₆ GHSP, currently being procured by DIL0 gmbh company (Germany). Being a plant for very peculiar and high challenging experimental activity, it can be expected that the SF₆ GHSP be operated relatively often

during the lifetime of the facility for maintenance and troubleshooting. Considering the above mentioned peculiarity of the installation (big size and need for rather frequent use), the adopted solution relied mostly on flexibility and cost effective off the shelf components to ensure reliability and maintainability, while the layouts of the distribution pipes and of the storage tanks have been customized to the requirements of the site installation, maintenance and operation of the MITICA facility.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 63 / 401

Plasma Control Requirements for Commercial Fusion Power Plants: A Quantitative Scenario Analysis with a Dynamic Fusion Power Plant Model

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One aspect of fusion power plant engineering with little insight is the plasma control for electric power generation. At the moment, no model can simulate the behavior of the fusion power plant from the plasma to the turbine generator. Whether the future fusion power plant operation would be load-limit or load-follow, or pulsed or steady state, it is very important to study the relation of the plasma behavior and the plant power output.

In order to obtain fundamental knowledge on the plasma control requirements for the future fusion power plant, the authors constructed a multi-domain dynamic simulation model of a nuclear fusion power plant on Modelica with Thermopower library. This is the first model of its kind that can simulate the power system behavior of a fusion power system.

The model plant was designed to have a tokamak reactor with solid breeder blanket cooled by super-critical water, with the output of 3,627 MWt (thermal) and 1,209 MWe (electric). A steam generator, two steam turbines (higher and lower pressure turbine), a turbine by-pass and a series of condensers were modeled as the power system (1st loop: 323/287 °C 6.85 MPa, 2nd loop: 274/81 °C 1.06 MPa). The power system was controlled by a closed-loop single-input-single-output (SISO) governor and a steam-regulating valve. The power system model was adjusted against the RELAP model of an existing PWR power plant.

The blanket and the diverter of the reactor were modeled as a transfer function of the neutron flux and the heat energy, based on experimental results by JAEA. This model simulates the thermal energy from the neutron flux as well as the decay heat of the system. It should be noted that plasma simulation is not within the scope of this study; the neutron flux was simply handled as an input signal.

The authors simulated several plasma patterns on this model. The plasma operation scenarios were created based on the load-limit and load-follow operation scenarios of SSTR; both short-term (< 10 sec) and long-term (~ 600 sec) fluctuations of the neutron flux were simulated.

Simulation results suggested requirements of the plasma operation in order to control the electric output within +/- 3% of the target for the first time, which is crucial for the commercialization of a nuclear fusion output.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 30 / 402

The influence of ELMs on low cycle fatigue behavior of ITER-like divertor target

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The high confinement mode have been considered as a major operation mode of ITER because of it's many advantages. However, in this case there will be severe shocks at the plasma boundary, called the edge localized modes (ELMs). It is significant and necessary to research the coupled mechanism of fatigue by both transient and periodic heat loads on PFC components. A FE model of typical monoblock divertor is established to investigate the coupled effects of ELMs on fatigue life. Different ELMs loads are coupled with the basic periodic high heat flux (HHF) loads, and the combined non-linear isotropic and kinematic hardening law was adopted to compute the thermal-mechanical effects. It is founded that: (1) Although the direct increase of damage induced by ELM transient loading is very limited in each cycle, the final damage increase at the end of HHF loading is times of previous value, indicating a nonlinear coupling effects of HHF loading and ELM transient loading. (2) Under the same HHF loading situation, The fatigue life time of target model is decreasing non-linearly with increase of ELM heat loads magnitude. (3) Under the same ELM loading case, with the increasing of HHF loading magnitude, The fatigue life time of first wall is decreasing more dramatically than previous low HHF loading magnitude, which means a more intensified coupled effects of damages.

Eligible for student paper award?:

Yes

T.OA1: Diagnostics and Instrumentation I / 403

Surface deterioration and recovery of CXRS first mirror in EAST

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First mirror (FM) is the key element of the optical and laser diagnostic systems in fusion devices such as ITER. Facing the plasma directly, it has to operate in an extremely harsh environment and suffer from the sputtering by high energy ions and charge exchange atoms, the impurity deposition due to wall conditioning and the sputtered wall materials etc [1,2], which results in the deterioration of the reflectivity and shorter lifetime. Protective shutter and plasma cleaning are widely studied to mitigate the deposition during the FM operating and remove the impurities deposits afterwards in recent years [3,4].

The non-plane large size (300 mm×80 mm×40 mm) FM, made of 316L SS was used for the charge exchange recombination system (CXRS), which has been operated in EAST for 361 days in three experimental campaigns from 2014 to 2016. The FM was exposed to the plasma with a total discharge pulse of 12499 shots and a total duration time of 86036 s. During the exposure, a capsule holder and

a motor shutter were used to mitigate the deposition. The scanning electron microscopy (SEM), electron energy dispersive spectroscopy (EDS), laser beam injection spectrum and a self-made laser system were used to characterize the surface morphology, impurity composition and the reflectivity of the CXRS FM. The inhomogeneous deposition consisting of C, O, Si, W and Mo was detected on the FM surface due to the shadow of the holder on FM surface and the gap between the shutter and the holder. The sizes of the particles were about several micrometers to tens of micrometers. Due to the severe deposition, the reflectivity of the FM was strongly decreased from 71% to 15% at the wavelength of 532 nm. To recover the FM surface and address the cleaning effectiveness and homogeneous of the non-plane large size mirror as well as find a possible way for FM in-situ cleaning in EAST in the future, the Ar plasma driven by the 13.56 MHz RF capacitively coupled system was used to clean the FM. After 187.3 h cleaning, the inhomogeneous deposition was visibly unseen and uniformly removed. The SEM and EDS results indicated that the micro morphology was developed during the cleaning and few residual particles consisting of C and O were also remained and covered about 5.3% of the cleaned mirror surface. The reflectivity of the FM surface was recovered to 68% which demonstrated the cleaning effectiveness and homogeneous of the RF plasma cleaning the non-plane large mirror. To prolong the FM lifetime, the optimization of the capsule holder and motor shutter was suggested and the regular in-situ cleaning was proposed.

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Eligible for student paper award?:

No

W.OA1: Materials II / 404

Experiemnt investigation on heat transfer performance enhancement of PFC hypervapotron by micro surface manipulation technology

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With the progress of ITER and the planned fusion-fission hybrid reactor, there is an urgent need to develop high effective and practical heat-transferring technologies for the removal of extremely high heat fluxes of about 20 MW m⁻². Hypervapotron, developed from a subcooled fin enhancement cooling concept, has received widely concerns. With the development of surface manipulation technology in recent years, the surface with micro/nano-structures was recognized as another pathway for boiling efficiency enhancement. This paper presents the efforts to enhance the subcooled boiling performance by a kind of micro surface manipulation technology, which produces 400×400μm pillars on boiling surfaces by combining photolithography with electroplating technology. Preliminary experiments present a significant increase in heat transfer performance as compared with plain test model, indicating a great potential to further enhancement of CHF.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 60 / 405

Analysis of Dogleg Duct Experiments with 14 MeV Neutron Source Using TRIPOLI-4 Monte Carlo Transport Code

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TRIPOLI-4 Monte Carlo transport code, developed by CEA, has been widely used on fission reactor physics and also can be used on fusion device neutronics. In order to verify the calculation features of TRIPOLI-4 code, a simple dogleg duct model was built to simulate the 14 MeV neutron transport based on a SINBAD fusion benchmark, called Dogleg Duct Streaming Experiment. The reaction rates in the bent duct and on the back surface of the experimental assembly for $^{93}\text{Nb}(n,2n)^{92}\text{mNb}$, $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ neutron activation dosimeters were calculated with the TRIPOLI-4 code. To improve the calculation efficiency, variance reduction techniques of TRIPOLI-4 were also performed. The calculation results showed that the variances reduction methods of the TRIPOLI-4 code are helpful, and obviously decrease the calculation time and increase the convergence efficiency. The calculation reaction rates results of 11 points inside and outside of the dogleg duct assembly were taken into account. Results from the TRIPOLI-4 simulation were compared with the experimental ones obtained from the measurements of FNS facility in Japan Atomic Energy Agency (JAEA). The benchmark results show that the TRIPOLI-4 code has a good potential to calculate and estimate neutron streaming effects in fusion device design.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 87 / 406

Impact of plasma configuration on impurity and density control during long pulse discharges in EAST

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ABSTRACT

As a fully superconducting tokamak device, the Experimental Advanced Superconducting Tokamak (EAST) bears the mission to demonstrate high-power, long-pulse plasma operation with flexible plasma configurations 1. In 2014, an ITER-like actively water-cooled tungsten (W) divertor was installed in EAST. The castellated monoblock structure in the W divertor facilitates the power handling up to 10 MW/m². Presently, the plasma-facing materials (PFMs) in EAST include tungsten

in the upper divertor, graphite in the lower divertor and molybdenum in the main vacuum chamber. The NBI shinerthrough region at inboard first wall and the guard limiter of heating antenna at outboard are also covered with graphite tiles. Last year, the long pulses over 100 s in L mode and 60 s in H mode have been successfully achieved at a loop voltage close to zero with the ITER-like actively water-cooled tungsten divertor [2]. The well-controlled plasma-wall interaction (PWI) in the edge plays a critical role in the long pulse discharge. The runaway growth of plasma density often terminates the discharge in a long-pulse operation, which is found to be connected with the edge recycling. The plasma shaping has an important impact on the PWI behavior, including particle exhaust, neutral recycling, impurity control etc. It is observed that too small gap between plasma and inner wall can cause intensive interaction and excessive heating of the graphite tile, enhancing erosion of carbon and overturning the particle balance, consequently influencing the density control. The plasma configuration in divertor is another important factor. Moving the strike point close to the pump slot between dome and outer target can increase the efficiency of particle exhaust significantly, benefiting the density control. Moreover, a good match of plasma shaping and the curve of guard limiter of heating antenna not only provides a good power coupling and current driving effect, but also is indispensable to reducing the impurity content in plasma. The optimized plasma shaping in the lithium-coated wall environment contributes greatly to the successful achievement of long pulse operation in EAST.

Keywords: lithium and silicon coating; tungsten erosion; impurity concentration; spectroscopic diagnosis.

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[2] Wan B.N. et al 2016 Overview of EAST Experiments on the Development of High-Performance Steady-State Scenario 26th IAEA FEC (Kyoto, Japan, 2016)

ACKNOWLEDGMENTS

This work was supported by the National Magnetic Confinement Fusion Science Program of China (Grant Nos. 2013GB105002, 2013GB107004 and 2015GB109001), as well as the National Natural Science Foundation of China (NSFC) (Grant Nos. 11575243, 11375010, 11605238 and 11575242).

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 17 / 407

High Priority Prototype Testing in support of System Level Design development of the ITER Radial Neutron Camera

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The paper describes the high priority testing activities supporting the ITER Radial Neutron Camera (RNC) design, performed by a consortium of European institutes within a framework contract placed by Fusion For Energy (F4E), the ITER European Domestic Agency.

The main role of the RNC is to measure the uncollided 14 MeV and 2.5 MeV neutrons from deuterium-tritium (DT) and deuterium-deuterium (DD) fusion reactions through an array of flux monitors/spectrometers located in collimated lines of sight (LOS) viewing the plasma through the ITER Equatorial Port Plug #1. The line-integrated neutron fluxes will be used to evaluate the radial profile of the neutrons emitted per unit time and volume (neutron emissivity) and therefore the neutron yield and the alpha particles birth profile.

The activity of high priority testing is dedicated to the preparation, the design of experimental test

environment, the conduction of appropriate tests and reporting of test results for the high priority prototypes, clarifying or verifying the expected key functions and system behaviour and enhancing learning on specific issue (potential showstopper).

The activities will focus on the development of experimental test rig, conduct and reporting of test for the high priority prototypes for the following items.

- Neutron Detectors and associated signal read-out equipment;
- Front-End Electronics & Algorithm development;

In addition, the Specific Grant 05 will dedicate effort for studying three conceptual options:

- the adjustable collimator concept
- the segmented detector concept
- environmental impact assessment on appropriate neutron detector technology (He-4, plastic and crystal scintillators)

The paper will also present the results of the first experimental results from the activities carried out in laboratory and in the irradiation facilities.

This work was partially funded by F4E under the Specific Grant F4E-FPA-327-SG05. The views and opinions expressed herein do not necessarily reflect those of F4E or the ITER Organization. F4E is not liable for any use that may be made of the information contained herein.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 49 / 408

Boron Carbide Coating on Tungsten By ICP Thermal Spraying

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Boron carbide was proved as a practicable material of in-situ protecting coating for tungsten tiles of Tokamak divertor, which is also expected to be presented towards the other plasma facing materials (PFM) in fusion device. In the work, B4C coating on tungsten substrates by means of inductively coupled plasma (ICP) thermal spraying technique is studied, which is driven by a 24-60 MHz RF power. Compared with arc plasma, ICP allows for considerable reduction of plasma contamination associated with electrode erosion. In order to investigate the effects of hydrogen introduced into sheath gas of ICP torch on B4C coatings fabrication, the characteristics of plasma at different Ar/H₂ proportion are diagnosed by optical emission spectroscopy (OES), and the melting processes of B4C particle in plasma are studied. For improvement of coating binding force, we prepared B4C/W coatings by the method of functionally gradient materials (FGM), in which B4C/W spraying powder was fed into ICP torch with gradient ratio. The characterization of the coatings are presented with compositional (XPS), structural (XRD) and morphological (SEM) analyses. And the testing under high heat loads and thermal cycling, together with the bonding strength is described. All the plasma properties and the characterization results of B4C/W coatings would give us an insight of improving the B4C coatings fabrication process.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 64 / 409

Study of a plasma boundary reconstruction method based on reflectometric measurements for control purposes

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In-vessel magnetic sensors will be drastically limited, if present at all, in DEMO, due to the pulse duration and the neutron flux, so alternative diagnostics systems based on in-vessel non-magnetic sensors and ex-vessel sensors are under investigation to meet the requirements of a safe and reliable operation of the machine. In particular, the signals provided by a microwave reflectometry system are being considered for application in the plasma position and shape control, as suggested by the results of preliminary experiments carried out in ASDEX-U device 1. In fact, local measurement of the distance between plasma boundary and first wall can be derived in correspondence of each reflectometric antenna. Indeed, reflectometric measurements are expected to be either missing or less reliable in some poloidal positions (top and divertor regions) and during transition phases (ramp-up, ramp-down) in DEMO. A crucial question to be answered is the minimum number of measurements and their poloidal position needed for a reliable plasma shape control. On the other hand, the knowledge of the whole plasma boundary allows a more accurate evaluation of geometric macroscopic quantities, such as plasma section, elongation, triangularity, which are of interest in the shape control and for an effective machine operation. Thus a plasma boundary reconstruction method, which uses only a limited number of local distances between plasma boundary and wall, was developed with the aim of contributing to answer the above mentioned questions. The plasma boundary is represented as a continuous curve to be reconstructed by applying an active contour technique. This approach has already been used in nuclear fusion research by exploiting the flux map provided by available magnetic measurements [2]. On the contrary, in DEMO case the boundary is reconstructed by deforming a curve so as to minimize a cost function given by the sum of the distances between each “real” boundary position derived by a reflectometric measurement and the currently estimated boundary position along the corresponding antenna line of sight. Among the proposed parametric contour models obtained as functional minimizing splines (“snakes”), B-splines were preferred for their compactness and because they implicitly force the curve smoothness. In fact, the resulting contour shape can be modified by just acting on a finite number of virtual control points related to the curve points through a time invariant matrix. The minimization of the cost function is an optimization problem, which was solved by a simulated annealing technique. Tests were carried out by assuming the number and positions of the antennas as foreseen in DEMO present design. A reliable reconstruction was achieved with the full set of 15 measurements except for the X-point region. The main equilibrium parameters were also computed and a satisfactory agreement was observed with the corresponding quantities of a DEMO reference equilibrium. Larger deviations were obtained with a reduced set of 10 measurements. Sensitivity of the results to measurement noise and antenna positions was also assessed.

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Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 50 / 411

The LIPAc Beam Dump

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The International Fusion Materials Irradiation Facility (IFMIF) aims to provide an accelerator-based, D-Li neutron source to produce high energy neutrons at sufficient intensity and irradiation volume for DEMO materials qualification. LIPAc is a 125 mA 9 MeV continuous wave deuteron accelerator whose components are under construction mainly in Europe, which is being installed in Rokkasho (Japan) with the purpose of validating the IFMIF accelerator design.

The beam is stopped in the interior of a copper cone (2.5 m long, opening angle 6.8 °), cooled by water flowing at high velocity along its outer surface. This piece is surrounded by a shield made of iron and low Z materials that attenuate the neutron and gamma radiation originated by the interaction of the deuterons with the copper. It incorporates dedicated diagnostics for beam dump monitoring: accelerometers to detect localized heating due to incorrect alignment of the beam and ionization chambers to detect changes of the beam shape outside the beam dump design limits.

One of the main difficulties of this beam dump is related to the fact that the interaction of the deuterons with the copper leads to the production of long lived isotopes (mainly Zn 65). A lead shutter has been designed to be inserted in the beam tube during beam-off periods to stop the gamma radiation escaping through the beam tube and allow access inside the accelerator vault. The joint of the beam dump to the beam tube has a special design that allows its remote disconnection at the end of life of the facility. As the cartridge activation precludes any maintenance activities in the beam dump and neighbouring elements downstream the lead shutter, a careful design and material selection has been done. The manufacturing is being performed following quality standards and performing strict acceptance tests.

The cooling water system includes a coil to delay the passage of the water from the accelerator vault to the heat exchanger room where most of its components are located, letting its activity to decay. pH, oxygen content and conductivity are controlled to minimize the corrosion of the copper cone.

This paper will describe the final design of the beam dump and related elements explaining the interrelations between them and the reasons behind their main features which in many cases have changed with respect to the first conceptual designs. It updates and completes previous publications providing detailed information of the validation tests performed or to be performed to the different components and their installation procedures.

Keywords: IFMIF, LIPAc, beam dump

Acknowledgments: This work has been supported by the Spanish Government in the frame of the Broader Approach Agreement (Spanish BOE n14, p. 1988) and also by project FIS2013-40860-R

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 78 / 412

Design and setup of the High Voltage Radio Frequency Test Facility for the characterization of the dielectric strength in vacuum of RF drivers for Neutral Beam Injectors Ion Sources

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PRIMA (Padova Research on ITER Megavolt Accelerator) is a large experimental facility under construction in Padova, Italy, aimed at the development and test of the full scale prototype of Neutral Beam Injectors (NBI), called MITICA, for ITER.

MITICA is designed to accelerate a beam of 40 A of negative deuterium ions up to 1 MV, in order to deliver a power of about 17 MW to the plasma with a pulse length of one hour; requirements that have never been reached before all together.

The negative ions are produced by means of an ion source composed of 8 radio frequency (RF) drivers working at 1 MHz, each generating a cold plasma at a pressure of 0.3 Pa with a power of 100 kW. A set of grids at different electrical potentials, extracts and accelerates the negative ions producing a negative ion beam which is then neutralized in order to enter and heat the plasma. To prove the possibility to achieve these requirements a second experiment will be hosted in PRIMA, called SPIDER: the full scale prototype of the ITER NBI negative ion source.

To gain experience on the RF voltage holding in vacuum, a dedicated experimental investigation is needed. Thus, the High Voltage Radio Frequency Test Facility (HVRFTF) is being built in Padova, at Consorzio RFX. The HVRFTF scope is to reproduce operating conditions of RF components in the ITER NBI ion source, in particular the voltage up to 15 kV at 1 MHz and the operating pressure in the range of 0.001 –0.3 Pa.

In HVRFTF a vacuum vessel is used to contain a low pressure atmosphere of the desired gas species produced by a gas injection and pumping system. The devices to be tested are placed inside and insulated from the vacuum vessel, and are supplied with RF voltage by means of a couple of feedthroughs. At first, circular planar stainless steel electrodes are used to derive Paschen curves with RF and dc voltage. The distance between the electrodes is adjustable, since one feedthrough is mounted on a bellow which is axially moved by a linear translator. Both the pressure and distance can independently adjusted, to derive the breakdown voltage threshold for the specific gas and for a specific configuration of electrodes. The RF high voltage is generated exploiting an LC resonance of a circuit supplied by a low voltage broadband amplifier. The operating frequency is adapted to look for the resonance frequency of the circuit, which is influenced by the parasitic capacitance of the device to be tested.

The paper will present the specific features and issues related to the design and setup of the HVRFTF and how they have been faced and solved.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 44 / 413

Challenges for the Wendelstein 7-X magnet systems during the next operation phase

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During the first operation phase OP1.1 of Wendelstein 7-X (W7-X) the magnet systems were not operated up to the maximum current. During the next operation phase OP1.2 the next step in the direction to a full current operation will be taken. Based on lessons learned during the first phase

the necessary improvements have been worked out to deal with the challenges in OP1.2.

The superconducting magnet system consists of the two different coil types: the non planar coils (NPC) generate the main magnetic stellarator field, whereas the planar coils (PLC) generate additional fields to provide the experimental flexibility of the device. With respect to OP1.1 the NPC current in OP1.2 will be increased slightly, but will be doubled in the PLC. Also a reversal of the current direction in the PLC will be required. This affects the electrical, the thermo hydraulic and the mechanical behavior. Test during and after OP1.1 showed that it might be advantageous to reduce the electrical stress during fast discharges. Therefore the dump resistor of the magnet protection system was optimized. It is customized for the next operation phase, but reduces the fast discharge voltage significantly from 2.7 kV to 1.8 kV.

In order to avoid the risk of a quench the magnet system is operated with a temperature margin of one K or more with respect to the critical temperature of the superconductor. A continuous comparison of the actual operation temperature with the maximum allowed temperature is being performed. In case of the safety margin being smaller than one K an automatic current ramp down should be initiated. During OP 1.1 a number of sensors did not work well which resulted in a blocking of the automatic function. In addition to check and repair of sensors and signal processing units the software was upgraded in a way that the operator is now able to exclude a sensor from the calculation also during operation.

The five trim coils are normal conducting coils mounted at the outer surface of the cryostat. They were operated during OP1.1 up to 1.1 kA which represents 2/3 of the maximum current. An important improvement resulted from the evaluation of one fast discharge of the superconducting magnet system. The trim coil power supplies detected an internal failure and switched off. This induced via the magnetic coupling a strong and fast voltage peak in the superconducting coils. The quench detection system interpreted this voltage as a sign of a quench and triggered a fast discharge of the superconducting system. For OP1.2 measures were studied and installed to minimize the cross link between the two coil systems.

The control coils are also normal conducting coils, but situated inside of the plasma vessel behind the divertor plates. These coils were commissioned for OP1.1, but not operated during the first plasma campaign. Nevertheless it turned out that the reliability of the power supplies needs to be improved, especially auxiliary systems like water cooling or driver boards of the inverter stages were sources of failures.

Eligible for student paper award?:

No

W.OA1: Materials II / 414

Integrating Materials Engineering and Design for Fusion

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The EU-DEMO fusion reactor is facing uncertainty in safety and licencing of in-vessel components that relate directly to the materials and design criteria used for assessments. These challenges come from the designed operation under unprecedented environmental conditions and reliance on the performance of complex in-vessel components over time spans of years. These critical components, including the divertor and breeding blanket, will rely on complex structures and multi-material interface that also utilise novel (in regards to nuclear environments) materials. Along with the high operational temperature and fusion spectrum irradiation effects the use of complex components provides key differences in the structural integrity and required design criteria for existing nuclear systems and the EU-DEMO reactor.

Many of the in-vessel components in DEMO have a high secondary (thermal) stress; this secondary stress can be relieved by geometrical changes that often occur in the plastic regime of materials deformation. The relaxation of stresses during plastic deformation of materials makes purely elastic analysis conservative and often makes design by analysis more challenging, potentially ruling out good designs. Most current nuclear design codes allow in-elastic analysis methodologies but have a preference to use elastic analysis with correction factors to accommodate plastic deformation, these corrections can add conservatism and inaccuracies to design. An overview of the use of in-elastic analysis methods for assessment of fusion components will be highlighted. The overview provides a representative example of a general requirement to move fusion design criteria towards alignment with modern assessment procedures that utilise in-elastic analysis as a precedence to improve design.

Overall, addressing the challenges in materials engineering and design criteria for fusion requires pragmatic adaptations and a new approach to the structural integrity case for fusion. The current pathways towards development of new DEMO specific design criteria and fusion structural integrity are reviewed.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 65 / 415

Design of 11 MA Snowflake divertor configurations of CFETR

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Chinese Fusion Engineering Test Reactor (CFETR) is under design, which will be operated in two phases 1. In phase I, CFETR is envisioned to provide 200 MW fusion power P_f and its designed main parameters are $R=6.6$ m, $a=1.8$ m, $BT=6-7$ T, $IP=10$ MA. However, in phase II which aims to DEMO validation, the P_f is over 1 GW and the IP increases to 11 MA. Considering the large P_f , it will be a serious challenge to handle the high exhaust power in scrape-off layer (SOL) of a single null divertor (SND) configuration, which means some solutions should be taken to reduce the heat loads on divertor plates. One solution is employing the snowflake (SF) divertor configuration.

In this paper, the capacity of poloidal field (PF) coils in obtaining the SN and the SF configurations was evaluated. The PF coils must remain their current limits. Instead of doing time-depending discharge evolution, static equilibrium analysis method was done to calculate the equilibria and the corresponding currents in PF coils at some fiducial points in a discharge by using TEQ equilibrium solver. The volt-seconds consumption for the ramp-up stage was estimated and then we calculated the PF coil currents of the SND and the SF configurations during the flattop phase with a range of I_i for the 11 MA H-mode inductive scenario. The results indicate that there is at least 100 volt-seconds of flattop for the SD, and the SF configurations cannot be established at the start of the flattop (SOF) because the currents in some PF coils exceed their limits. By adjusting turns of the PF coils, all three kinds of SF configurations (SF plus, exact SF, SF minus) were realized at least after the SOF point. The properties of the SF were also analyzed. The connection length, flux expansion of the SF all significantly increased by at least 1.5 times over the SND.

Keywords: CFETR, PF coil, equilibrium, snowflake

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1 Yuanxi Wan et al 2016 26th IAEA FEC, Paper No. OV/3-4

Eligible for student paper award?:

Yes

W.OP2: Heating and Current Drive / 416

Towards a new generation of high power high efficiency neutral beam heating system for the future fusion reactors

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Since the signature of the ITER treaty in 2006, a new research programme targeting the emergence of a new generation of Neutral Beam (NB) systems for future fusion reactors has been underway at CEA, several academic laboratories in France and EPFL (Switzerland). To provide plasma heating and current drive, the NB system specifications are very demanding: a very high level of neutral power (up to 150 MW) and energy (1 MeV), including high wall-plug efficiency ($\eta > 60\%$), high availability and reliability. To meet these specifications, a novel NB concept based on the photo-detachment of the energetic negative ion beam is under study. The keystone of this new concept is the achievement of a photo-neutralizer where a high power photon flux (~ 3 MW) generated within a Fabry Perot cavity will overlap, cross and partially photo-detach a narrow and tall "blade-like" negative ion beam accelerated to high energy (1 MeV). It will be shown that such a photoneutralization based NB system would have the capability to provide several tens of MW of D₀ per beam line with a wall-plug efficiency higher than 70%. The talk will describe the injector concept and the main achievements in 2016; in particular, to provide a blade-like ion beam, a first Helicon plasma jet of 1.8 meter long in hydrogen with a density $n \sim 5 \cdot 10^{17} \text{ m}^{-3}$ has been achieved at EPFL. At the LAC laboratory, a reduced scale photoneutralization experiment has demonstrated 50% photoneutralization in continuous wave on a 1.2 keV H⁻ beam under 10 kW photon power stored in an optical cavity.

Eligible for student paper award?:

No

M.OP2: Materials I / 417

Thermomechanical properties of nanostructured W based coatings under ITER-relevant thermal loads

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The full tungsten (W) divertor of ITER will suffer from extreme thermal loads during both steady and transient operating conditions. These thermal loads, together with energetic species bombardment from the plasma, induce W surface erosion, melting and recrystallization. The sputtered particles

could migrate and form micrometric thick co/re-deposits on the peripheral regions of the divertor. These layers, together with recrystallized and solidified W, show different thermophysical properties from bulk W and must be opportunely characterized.

Since we actually are not able to recreate the full ITER environment, it is mandatory to study at the lab-scale how W based components behaves in ITER-like operating conditions. In order to study the behavior of co/re-deposits expected in ITER, in previous works we exploited the versatility of Pulsed Laser Deposition (PLD) to deposit W based coatings, namely pure W coatings with tuned nanostructure and morphology and W-oxide coatings with different oxygen contents. These coatings, chosen as proxy of co/re-deposited W, have been exposed to ITER-relevant plasmas, and their deuterium retention properties, as well as their structural and morphological modifications after exposure, have been assessed [1, 2].

In this work we characterize the mechanical properties (i.e. stiffness and ductility) of these PLD W coatings by Brillouin spectroscopy (BS), as function of nanostructure (e.g. crystallite size) and oxygen content. Thermal properties, i.e. coefficient of thermal expansion, are also assessed by an ad-hoc developed experimental setup based on substrate curvature measurement. In addition, we investigate their thermomechanical behavior under two different scenarios that mimic steady and transient ITER operating conditions. For the former case, we perform standard thermal annealing treatments at 200–1000°C on nanocrystalline-W (nano-W) samples, in order to study their behavior at ITER-relevant steady operating temperatures. We focus on the mechanical, structural and morphological properties modifications upon heating. BS is exploited to derive the mechanical properties, while samples structure is assessed by SEM and XRD analysis. We find that nano-W starts to crystallize at around 600 °C, which is well below the bulk W recrystallization temperature (i.e.1400°C); at this temperature, comparing to as-deposited nano-W, an increase by 60% of material stiffness with a corresponding loss of ductility by 30% is observed [4]. In addition, we expose the as-deposited coatings to nanoseconds laser irradiation. Nanoseconds lasers have been already exploited for mimicking thermal effects induced by ITER-like transient events (e.g. disruptions, ELMs)[3]. Here, exploiting the same Nd:YAG laser system we used for PLD, we look for thermal effects (e.g. cracks formation, melting) as function of laser energy fluence. The characterization is assessed by SEM morphological analysis. The fluence thresholds for the thermal effects are then compared with the ones obtained by the irradiation of bulk W plates, selected with different surface finishes. The measured experimental thresholds are compared to the ones obtained by numerical simulations using a 2D thermo-elastic code developed to this purpose.

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Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 18 / 418

Developing the Simulation of Spectra Code Based on HL-2A tokamak Motional Stark Effect and Beam Emission Spectroscopy

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Developing the Simulation of Spectra Code Based on HL-2A tokamak Motional Stark Effect and Beam Emission Spectroscopy

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Abstract: Beam emission spectroscopy (BES) diagnostic system had been developed on HL-2A tokamak. Combined with motional stark effect (MSE) diagnostics, it could get radial electric field E_r and safety factor q , which play an important role on the plasma-control and impurity transport process. The simulation of spectra (SOS) code developed by TU/e fusion group was used for spectra simulation based on various tokamak device real situation, which used its tokamak parameters. It could be used for designing spectral diagnosis system and prediction performance of its characters. In this paper, based on HL-2A experiment running condition parameters, simulated BES and MSE spectra by SOS were consistent with experimental data fitting results. It also got calculated magnetic field pitch angle and other parameters via experimental data, which also agree with the results of simulation code. The results reveal that the SOS code can be used to design similar diagnostic systems for its reliable prediction in the future ITER.

Keywords: NBI diagnostics, Beam emission spectroscopy, Motional stark effect, Simulation of spectra
 Acknowledgments: This work was supported by the members of HL-2A team. The author will be grateful to Von Hellermann for constructive discussion.

Eligible for student paper award?:

Yes

W.OP3: Blankets and Tritium Breeding: Solid Breeders / 419

Study of the pebble beds for tritium breeding blanket

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The solid blanket is a candidate tritium breeding blanket concept for Chinese Fusion Engineering Test Reactor (CFETR), and the thermal and mechanical characterization of the ceramic pebble beds are vital for researchers to know for a reliable solid blanket design. Some related numerical and experimental studies were conducted at University of Science and Technology of China and China Academy of Engineering Physics, etc. in China. A theoretical model, coupling the contact areas with bed strains, was developed to predict the effective thermal conductivity of mono-sized ceramic pebble bed. The influences of parameters such as properties of pebbles and gas, bed porosity, pebble size, gas flow, contact area, thermal radiation, contact resistance are all taken into account in this model. Besides, the behaviors of granular materials under mechanical cycling were investigated, including the effect of mechanical cycling on the granular compaction and the evolutions of the elastic modulus and force chains with the discrete element method (DEM) method. DEM-CFD simulation of purge gas flow characteristics in a solid breeder pebble bed was studied also. Furthermore, two experimental platforms using transient thermal probe method and transient plane source method respectively were built and operated to measure the effective thermal conductivity of the pebble beds. Li_4SiO_4 and Li_2TiO_3 as promising candidate tritium breeder materials have been considered. The measured temperature ranged from RT to 600°C, and the helium or air at 1 bar was used as filling gas. The experimental results were obtained and some phenomena were discussed.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 61 / 421

Application of Gaussian Processes for predicting tritium breeding ratio in the helium cooled pebble bed breeder blanket

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Assuring self-sufficient tritium production is of critical importance to prospective Deuterium Tritium (DT) fusion power plants.

Achieving a tritium breeding ratio in excess of 1.1 has been identified as a key requirement for future DT fusion power plants.

Tritium breeding ratio values are typically calculated via computationally expensive neutronics simulations as an integral stage in the development of breeder blanket designs.

This paper reports on an application of machine learning which is able quickly predict tritium breeding ratios for different variants of the helium cooled pebble bed breeder blanket.

Previous research in this area has used simplified 1D and 2D models to simulate a broad range of neutronics parameters for variants of the helium cooled lithium lead \cite{Jaboulay20132336}; machine learning (neural networks) was then used to make predictions.

In this research, detailed 3D neutronics fusion reactor models were simulated to find tritium breeding ratio values.

Breeder blankets with different neutron multiplier pebble bed heights, lithium ceramic pebble bed heights, lithium-6 enrichments and neutron multiplier materials were created as part of the model.

Training data was formed from these input parameters along with the resulting tritium breeding ratio for 2200 simulations.

A form of machine learning (Gaussian Processing) was then applied to the training data using the scikit-learn Python library.

The result of this research is a predictive function that is able to provide accurate predictions for the tritium breeding ratio for a range of different blanket designs.

Additionally any prediction of tritium breeding ratio also returns the associated standard deviation. In cases when the returned standard deviation is too large the system is able to perform an automated neutronics simulation which calculates the tritium breeding ratio at the point of interest.

Resampling techniques (jackknifing and cross sampling) were performed on the training data to estimate the variance of the estimator and the error involved in making predictions across the dataset.

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 67 / 422

INTEGRATION OF METALLIC SEALS ON CIRCULAR FLANGES FOR NEUTRAL BEAM FRONT END COMPONENTS

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The Neutral Beam (NB) system for ITER is composed of two heating neutral beam injectors (HNBS) and a diagnostic neutral beam injector (DNB). A third HNB can be installed as a future up-grade.

This paper will present the design solution of the sealed interface between the components so called 'Neutral Beam Front End Components'. The components to be considered are the Drift Duct, the Vacuum Vessel Pressure Suppression System box (VVPSS box), the Absolute Valve, and the Fast Shutter. These components connect the Neutral Beam vessels of the injectors to the Tokamak Vacuum Vessel. These components are connected with circular flanges bolted together. They are all first confinement barrier and by the way Safety Important Components classified. They must comply with stringent requirements in term of leak tightness and robustness. They are all classified RH class 3 that means that their life time shall comply with ITER life time without planned maintenance. In case of unlikely incidents or accidents, and regarding the results of Neutronic analysis in the Neutral Beam cell, the safety approach is to consider that all operations will be done fully remotely. It is not likely that it will be acceptable to allow human intervention.

The paper will describe the design of the interface solutions which have to be implemented between these components regarding the primary vacuum confinement and the remote maintenance operations. The IO baseline solution to ensure the confinement and the leak tightness at these interfaces was the lip seal weld. The design and manufacturing code chosen for the NB FEC is the RCC-MR. The Lip seal weld solution raised two main concerns which are the compliance with the RCC-MR code and the feasibility of the full remote maintenance operations. The cutting and re-welding operations of the lip seal weld have never been demonstrated without a human intervention. And the RCC-MR is not directly applicable (or not relevant) for the lip seal weld. It appears that the design of the lip seal weld raises a lot of open issues like feasibility, tests and full RH operations inside the NB cell. The solution with two metallic seals with a pumped interspace will solve all lip seal weld concerns. A complete study has been carried out to demonstrate the compliance of this solution in term of leak tightness requirement, feasibility of a full remote maintenance operations and improvements of testing and monitoring these key interfaces.

Eligible for student paper award?:

No

M.OA1: Experimental Devices I / 423

MAST Upgrade Divertor Facility: A test bed for novel divertor solutions

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The challenge of integrated exhaust consistent with the other requirements in DEMO-class tokamaks (ITER-like and alternative DEMOs, FNSF approaches) is well-known. The exhaust solution is likely to be fundamental to the design and operating scenarios chosen. While no facility can address all of the challenges, the new MAST Upgrade tokamak can explore a wide range of the aspects related to the divertor plasma. MAST Upgrade has unique capabilities to produce conventional and novel divertor configurations for detailed studies and comparison in a single device. The two closed divertor chambers are each surrounded by 8 poloidal field coils for detailed control of the magnetic geometry, including strike point location, field line length within the divertor, poloidal flux expansion and their variation across the scrape-off layer, whilst keeping the shape of the core plasma unchanged. It will be equipped with neutral beam heating, and a wide range of high resolution diagnostics with a strong emphasis on the scrape-off layer and divertor plasma, allowing new levels of detail in testing of models.

To extrapolate to future devices where full tests in advance are not feasible, theory-based and semi-empirical models can be used. These models, and their necessary compromises and simplifications,

need to be validated and improved using the plasma physics mechanisms expected to be important at DEMO-scale, and this is at the heart of the MAST Upgrade programme. Possible paths to confident performance predictions will be outlined, with the role of MAST Upgrade indicated.

Specific physics areas to be explored include:

- i) Plasma detachment, especially how novel magnetic configurations can make detachment easier and more controllable, e.g. the role of variation in mod(B) along the divertor leg.
- ii) How divertor configuration and detachment state affect the plasma pedestal and access to H-mode.
- iii) Controllability of double null, with potentially different detachment behaviour in upper and lower divertors.
- iv) Behaviour of the inner leg in double null for different configurations (SX, SF, conventional).
- v) Cross-field transport which determines the power footprint on the divertor and the ease of detachment. Longer divertor legs allow cross-field transport to be more effective.

In most cases the studies will focus on the underlying mechanisms, e.g. plasma filaments / blobs are often actors, and are affected by the scenarios and divertor configurations and state of detachment.

While MAST Upgrade is not a prototype, in this presentation we will address how it can be used to inform design questions for alternative and novel divertors in DEMO-class devices.

This work has been funded by the RCUK Energy Programme [grant number EP/I501045].

Eligible for student paper award?:

No

T.OA2: Divertors and PFCs: Tungsten / 424

TUNGSTEN TECHNOLOGY DEVELOPMENT IN KOREA AND ITS APPLICATION TO KSTAR EXPERIMENTS

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Main focus of fusion engineering has been moved towards development of metal wall plasma facing components (PFCs) and corresponding interaction between plasma and metal wall. National Fusion Research Institute (NFRI) has started metal wall related research activities since 2012, which are closely related to major upgrade of KSTAR and research on K-DEMO. As the first step, metal bonding

technology has been developed and tungsten brazed block samples with good bonding quality have been obtained. Bonding technology for tungsten monoblocks using HIP is currently under development. Two major issues on tungsten divertors with castellated structure in ITER and beyond, are steady state and transient power handling capability and fuel retention inside the gap. Monoblocks aligned perfectly to their neighbors have leading edges directly exposed to plasmas. Leading edges under high power ELM H-mode can be melted in several seconds of plasma exposure time. In order to solve this issue, radiated divertor and shaping of castellated monoblocks are proposed: Optimization of the shape and the angle of the castellation structure can reduce significant amount of heat load on the PFCs. Tore Supra has found that fuel retention was dominated by co-deposition, especially at the gaps of tile blocks. In order to study those two issues, special tungsten block tiles with various shapes of castellated structure with leading edges were fabricated and installed on the central divertor of KSTAR. It is found that the leading edge heat load can be described by using simple optical approach without Larmor orbit effect. Results also indicate clearly that the shape-optimized block has more heat load handling capability compared with conventional one, and the maximum temperature under heat load is much lower. The contributions of ions and charge-exchange neutrals on the deposition inside the gap of various shapes and heights of castellation structures have been measured and a complete set of deposition profiles inside the gaps was obtained. 0.3 mm misalignment allowed in ITER shows no meaningful difference on deposition profile. Since KSTAR has not enough heating power to sputter tungsten atoms from the blocks, transport of tungsten atoms in plasmas cannot be studied. We have developed a gun type powder injector to put tungsten powders into L- and H-mode plasmas, which provides evaporation of tungsten powders releasing a large amount of tungsten atoms. Vacuum Ultra Violet (VUV) spectroscopy, whose wavelength is around 6 or 12 nm, is used for the tungsten line measurement. Tungsten powder injection experiment has been successfully performed and the core accumulation of tungsten atoms is measured. Obtained tungsten emission spectra show very similar features measured at ASDEX Upgrade indicating tungsten atoms were evaporated from the powder and penetrated into the core. This has opened a new research area in KSTAR despite of low heating power.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 19 / 425

First measurement of LiIII charge exchange line on EAST tokamak

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Lithium wall conditioning on EAST have been employed since 2009. The high performance plasma (H-mode) has been recently successfully obtained with the help of lithium coating since the autumn campaign of 20101. And stationary H-mode plasmas over 30s was obtained in 2012. Lithium evaporators and real-time Li power/granules injection are used for the stationary H-mode plasma and ELMs control. One neutral beam injection (NBI) with two positive ions sources positioned at the A port has been successfully installed and running on EAST in 2014[2]. It injects a beam of deuterium atoms with an energy of 50-80keV and a power of about 1-2MW to heat and rotate the plasma. A toroidal charge exchange recombination spectroscopy (CXRS) based on the beam of A-port is developed and installed on EAST [3] at the same time. It uses a high-throughput, lens-based scanning spectrometer which can be adjusted to any wavelength between 400 and 700nm and a back-illuminated frame-transfer CCD camera with on-chip multiplication gain.

On EAST, the active charge exchange lithium emission ($\lambda=516.689\text{nm}$, $n=7-5$ [4]) is firstly observed during the 2016 experimental campaign on EAST. The simultaneous measurement of CVI and Li III was performed by dropping lithium power at the same time during the 2016 EAST experimental

campaign. The preliminary measurements suggest there are sufficient lithium emissions to allow for the measurement of lithium spectra when the real-time Li power is injected. In the paper, the experimental hardware is described and preliminary measurements will be shown.

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Eligible for student paper award?:

No

W.OA3: Neutronics and Multiphysics Analysis / 426

Development and Validation of Cryostat Finite Element Model with Unique FE Method

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The ITER Cryostat—the largest stainless steel vacuum pressure chamber ever built which provides the vacuum environment for components operating in the range from 4.5k to 80k like ITER vacuum vessel and the superconducting magnets. The Cryostat being a Safety Important Class component, Design validation at each stage is necessary if there is any deviation. The Cryostat is currently at manufacturing stage, frequent changes major/minor are coming during manufacturing process, some of which being judgmental needs fine assessment. Assessing the effect of these changes using the conventional FE method needs significant time and effort. Also the need of iteration for every change further increases the time and effort by manifold in just making FE model. This paper will present a unique method to develop FE model of complex systems like Cryostat and enables to validate the structural strength of the system during any load or load combinations. One can incorporate frequent changes quickly and assess with ease. This paper mainly focuses on the details of the different approach in development of FE model of Cryostat from current manufacturing Model of Cryostat. In this method the context and complexities identified and component wise thirty FE models of Cryostat are generated. Then these FE models are integrated into a full Cryostat (assembly) FE model using suitable contact definitions. This approach facilitates to incorporate component level changes without affecting the whole Cryostat FE model, thus saving in time and efforts of re-creating the meshed model. Secondly it demonstrates simplification in Cryostat Bearing model [3,4]. Lastly validation of the analysis result of this FE model with ASME Section VIII Div 2 and with previous Cryostat assessment [1,2]. This approach once developed, reduce time and effort drastically which makes iterations easier and hence enables quick decision making for the Design Responsible authority.

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Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 51 / 427

Profile Tolerances influence on Cryostat Base Section

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The ITER cryostat—the largest stainless steel vacuum pressure chamber ever built which provides the vacuum environment for components operating in the range from 4.5k to 80k like ITER vacuum vessel and the superconducting magnets. The Cryostat is divided into four section, of which, Base section is most complex because of its web shaped structure sandwiched between two 60mm thick plates with stringent requirements in manufacturing tolerances. The required profile tolerance is 30(+10/-20) mm at weld locations and 20(+/-10) mm at other locations but during manufacturing, the tolerances are observed to be 60mm (+10/-50) mm at weld location and 50mm (+10/-40) at other locations. This increased profile tolerances are expected to affect the structural behaviour of Cryostat. The present paper discusses assessment of these tolerances on Cryostat Base Section using FEM software, Ansys. The increased profile tolerances on Base Section are applied using two different methods [2]. Maximum tolerance value was considered and five cases were identified to the complete effect of increased profile tolerances on Base section. Limit Load method 1 is used to analyze structural impact of these increased tolerances cases on Cryostat Base Section. The impacts of critical tolerances were discussed in the paper.

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Eligible for student paper award?:

No

M.OP1: Plasma Operation and Control / 428

Real-time control of MHD instabilities using ECCD

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For ITER and also future devices like DEMO, the capability to control magneto-hydrodynamic (MHD) instabilities like sawteeth and neoclassical tearing modes (NTM) is required to ensure reliable high β plasma operation. Such plasmas naturally encounter these phenomena, some of which are performance limiting at best but destructive in the worst case as NTMs may cause disruptions. A tried and trusted actuator with which these instabilities can be strongly influenced and eventually controlled is electron cyclotron resonance heating and current drive.

ASDEX Upgrade is making a large effort to develop, operate and evaluate an ECCD based, generic solution to MHD control, easily portable to new devices like ITER. Depending on the control strategy with highest priority, be it sawteeth or NTMs, ECCD deposition may need to be targeted at different radial locations, but the general control scheme is very similar, hence a so-called supervisory controller can delegate tasks to lower-level controllers and achieve a globally ideal solution given the existing constraints. Moreover, thanks to its generality, it can easily be adopted by other plasma experiments. TCV, among others, has started similar programs.

In order to achieve precise deposition control, a large number of real-time diagnostics and intelligent controllers work in unison, all coordinated by the discharge control system (DCS). In addition to the real-time equilibrium reconstruction, which is essential for our application, we require density profile measurements, real-time detection of MHD marker positions (rational surfaces, inversion radius, etc.) and some global plasma parameters (I_p , β_{pol}) which complement the dataset on which the controllers base their decisions.

Using the system in closed loop operation with 4 completely independent actuators, we have achieved controlled stabilization of 3/2 NTMs at β_N of 1.8 and preemption of NTM onset using the same control mode reaching β_N of 2.3 without mode. The system can automatically identify magnetic islands and aim stabilizing ECCD at the appropriate rational surfaces. Newly introduced deposition sweeping schemes alleviate the deposition accuracy requirement for NTM stabilization such that even imperfectly measured flux surface geometry is not prohibitive for achieving the intended goal.

The controller approaches maturity and is undergoing optimizations to improve its performance and reliability. For this step, we employ detailed data analysis with a beam tracing code to determine the physical limits of successful stabilization. In the case of NTMs these are dependent on the ratio of externally driven current to bootstrap current at the location of the magnetic island. Detailed analysis and a full system overview are being presented.

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

Eligible for student paper award?:

No

W.OP1: Magnets / 429

Commissioning of the Wendelstein 7-X In Vessel Control Coils

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The magnet system of the stellarator fusion device Wendelstein 7-X (W7-X) is composed of three different groups of coil systems. The main magnetic field is created by a superconducting magnet system that is accompanied by two sets of normal conducting coil groups, the trim coils positioned outside of the cryostat and the control coils inside the plasma vessel.

The control coil system consists of ten 3D shaped coils, cooling systems, high current feeds and an autonomous remote control system. The design of the ten individual power supplies is based on four-quadrant current converters using Mosfet-Transistors. They provide individual bipolar DC currents and a superposed common AC current of low frequencies for each coil. The coils are situated behind the baffle plates of the ten divertor units. They are fed with electrical power and cooling water by a plug in that is also sealing the vacuum conditions inside the plasma vessel from the ambient outside the machine. The magnetic field created by the control coils system allow for the correction of error fields, to influence the islands at the plasma boundary and for the sweeping of the separatrix, e.g. the point of the largest power position, across the divertor.

At the end of 2015 the installation of the control coil system was completed and the integral commissioning took place in parallel to the ongoing completion of W7-X. For the first time the control coils and their power supply were operated in conjunction with all auxiliary systems like the power distribution system, the high current feeds, the cooling system and the safety control system.

This paper describes the results obtained and experiences made during the integral commissioning of the control coil system, including the baking process in preparation for the first experimental campaign of W7-X.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 20 / 430

Preliminary Design of Laser-Induced Breakdown Spectroscopy Diagnostic for Divertor Analysis in EAST

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Preliminary Design of Laser-Induced Breakdown Spectroscopy Diagnostic for Divertor Analysis in EAST

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Analysis and understanding of plasma wall interaction which results in deposition, erosion, and fuel retention on the plasma facing materials (PFMs) is among the most important task for magnetic confinement nuclear fusion devices. Laser-Induced Breakdown Spectroscopy (LIBS) is a well-established elemental composition analysis method as well as one of the most promising method for the wall diagnosis of fusion devices in situ. A LIBS system has been developed and applied in situ to measure and monitor the composition evolution on the PFMs at the high field side of superconducting and long-pulsed EAST tokamak starting from the 2014 campaign. LIBS signal provided the fuel, impurities and lithium erosion/deposition distribution in real time between main plasma discharges and during wall conditioning processes [2]. The result shows that H, D, Li, Mo, C, W, Si, et al. elements lines can be identified from the LIBS spectra. The lithium-deuterium and -hydrogen co-deposition layers with thickness between few hundreds of nanometres and few micrometres can be found on the first wall of deposition area. The H/(D+H) ratio on the first wall was obtained between 20%-30% which is much higher than in the main plasma. In this work, a conceptual study for an upgraded LIBS system at the “H port” of EAST tokamak will be presented. The detected area of the system will cover the divertor region of EAST. The LIBS spatial resolution on the divertor tile surface is able to reach to 0.2 mm to present 2D mappings of each element in both erosion and deposition areas by using a series of motorized optical component integrated in an endoscope device. The spectral wavelength range of 200-900 nm with resolution of 0.1 nm and maximum laser energy of 330 mJ@1064 nm with 5 ns pulse width can be achieved by using a multichannel spectrometer and a Nd:YAG laser, respectively. An ICCD detector and a ps pulsed laser system will be installed in future to further improve the spectral sensitive and ablation depth resolution on materials. This work can be valuable for the understanding of plasma wall interaction about divertor physics in long-pulsed fusion device.

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Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 108 / 431

Inspection Method for Delamination Defect in First Wall Panel of Tokamak Device by using Laser Infrared Thermography Technique

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First wall panels (FWP), which adjoins along the inner wall of the vacuum vessel (VV) of Tokamak device, are multilayer structures different materials welded by solid welding technique to perform heat exchange enhancement, VV protection and tritium breeding functions. In order to implement online inspection of the delamination defect of FWPs nondestructively, a NDT method capable to make inspection without accessing into the VV (through the VV window) is necessary. In this paper, a laser infrared thermography testing (LIRT) method is proposed to deal with this problem considering its features of remote sensing, non-contacting and high detection efficiency. To check its feasibility under practical inspection environment, several inspection modes are considered based on the practical structure and size of FWP and VV in EAST Tokamak device. Different distances and angles of FWPs to the LIRT transducers are considered to investigate its detectability for FWPs at different positions. A laser infrared thermography testing experiment system is established and several double-layered plates with different artificial delamination defects are inspected according to

the selected testing conditions. In addition, several image processing methods including pulse phase method (PPT), principle component thermography (PCT) and thermography signal reconstruction (TSR) are adopted to enhance the detectability. In this way, the feasibility of the LIRT method for inspection of delamination in FWPs is clarified. Finally, In addition, an online inspection procedure for delamination defects in FWPs through VV windows in practical condition is presented.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 80 / 432

Structural and thermal analysis of a distributed ICRF antenna integrated in European DEMO blanket

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The use of efficient heating and current drive systems is an important research priority for DEMO. One such system in consideration for the European DEMO is the ion cyclotron range of frequency (ICRF) heating system. Extensive operational experience on several existing fusion facilities, a relatively low cost and high plug to power efficiencies motivates the consideration of ICRF for the heating and current drive mix in the European DEMO project.

In the present baseline configuration considered for DEMO, the ICRF antenna consists of several radiating metallic straps integrated in the blanket's first wall, and protected at the front, if necessary, by a toroidally segmented structure called the Faraday shield. Since the antenna shall be integrated in DEMO, a main requirement is its ability to withstand the stringent operational conditions, including plasma steady state and transient loads. Thus, the antenna design shall be designed from the start to be compatible with such loading conditions. Furthermore, simplifying the overall machine design is naturally desirable. Since the antenna lifetime has to be equivalent to that of the blanket in terms of neutron fluence the choice of the structural material is currently limited to EUROFER.

This paper provides a definition of the most critical (i.e. thermo-mechanical) loads to be considered when creating the ICRF antenna design. The structural integrity of the baseline ICRF antenna configuration is then analyzed using finite element analysis tools against the given primary loads in DEMO. The results of this exercise will be reported in this paper. Furthermore, the baseline antenna design is modified where ever necessary to adapt the operational loads in DEMO. Any design change will ultimately need to fulfill the DEMO machine constraints and RF physics requirements. Results presented here may thus not be final, and further iteration might prove necessary.

Presently, four blanket concepts are being considered for DEMO, though two cooling liquids are used: Helium and water, for all four blanket types. Whenever possible, cooling of the ICRF antenna shall be part of its hosting unit, i.e. the blanket. This particular issue has been considered during the investigation as described in this paper.

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

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 31 / 433

Preliminary progress of the divertor module in CFETR system code

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To develop the concept of China Fusion Engineering Test Reactor (CFETR), a system code for integrated simulation and optimization is being developed. It is a platform which provides the tools of Tokamak conceptual design and engineering analysis. Meanwhile, the detailed design models and corresponding analysis results would be stored in the data management system of this platform. The relevant data can be shared with the other modules in the platform. It contains information about constrains for each individual module and co-constrains among different modules, such as heat flux, electromagnetic force, spatial position and so on. In order to meet all requirements of every module and simplify the optimization processes of the design models, design workflows and engineering principles are set in the development of the system code. Therefore, every authorized engineer could get the up-to-date data associated with his own design module and submit the approved research and design result in this platform. It keeps the data synchronization update in the collaborative design of a fusion reactor for all the designers. This paper presents the development progress of the divertor module in CFETR system code and mainly focuses on normalization of the whole design workflow of the Tokamak divertor design and the correlative technical solutions. For the description of the CFETR system code diverter module, a simply design and analysis workflow also be presented.

Eligible for student paper award?:

Yes

T.OP2: Fueling, Exhaust, and Vacuum Systems / 434

Hydrogen Isotope Separation by Cryogenic Chromatography in Processing Tokamak Exhaust Gas

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In addition to rapid recovery and processing plasma ash discharge gases, fast separation of hydrogen isotopes and rapid D-T fuel balance is also an important technical content of TEP system. In this paper a cryogenic chromatography method for reprocessing Tokamak exhaust gas is described. The experimental apparatus consists of a column with carbon molecular sieve used as exhaust storing and purification unit, four columns with 5A molecular sieve operated at the temperature liquid nitrogen to adsorb protium, deuterium and tritium. In order to raise the efficiency and to shorten the total time of isotope separation and the total length of columns between two columns, a disproportionate equilibrators for isotope exchange of HD and HT was inserted. After passing through the cascade columns, the protium, deuterium and tritium are separated cleanly. The overall recoveries of deuterium and tritium for cleanup isotope separation procedure are greater than 97%, the losses are less than 5%, and the protium removing efficiency is larger than 98%. Further refining the process parameters, be able to do better as a result, and is expected to meet the requirements of self-sustaining tritium fuel cycle.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 52 / 435

Irradiation effects on lifetime of first wall structure materials for CFETR

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China Fusion Engineering Test Reactor (CFETR), designed as a bridge connecting ITER and DEMO, was proposed to achieve long-term stable operation with 30–50% duty time factor at low fusion power (50–200 MW). First wall of CFETR services on the conditions with high surface heat flux and intense neutron irradiation. The existing structural design rules for first wall mainly involved stability analysis, which neglected the primary failure mechanisms related to service time such as creep, fatigue and irradiation effects. With the approaching of CFETR engineering, it is important to analyze the effects of swelling, embrittlement, irradiation creep, and irradiation fatigue on the lifetime of the structure materials. It is the main purpose of this work, based on the existed neutron irradiation data of three kinds of candidate structure materials (ferrite/martensite steel, austenite steel and oxide dispersion strengthened steel) and finite element simulation on the service conditions of first wall for helium cooling solid breeding blanket.

Since the maximum irradiation doses in CFETR are 10dpa and 50dpa in lifetime of designed phase-I and phase-II respectively, according to the development roadmap of nuclear fusion energy in China, so the irradiation swelling would not be the most important issue compared to the creep for long service time requirement. The irradiation creep lifetime was evaluated with Larson-Miller Parameter model, and the irradiation fatigue lifetime was predicted with S-N curves. The allowable irradiation creep lifetime decreases with increasing of surface heat flux (0.3-0.7 MW•m⁻²), first wall thickness (1-5 mm) and inlet coolant temperature (300-500 °C). For the current CFETR conceptual design condition, the lifetime is not limited by irradiation creep, which indicated the room for lifetime improvement and design parameters optimization.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 104 / 436

Concept of the integrated environment of management by large scientific projects

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The developed concept has long history of the development. The first work on this subject was published in 1993 1. Initially the concept developed as the tool of the system analysis of safety of the ITER reactor project, at this time on the basis of the conceptual project of the ITER reactor was developed multilevel (up to the 19th level of hierarchy) structurally functional hierarchical computer model of the ITER reactor. The hierarchical structural functional model was developed on the basis of constructive indications. It means that the principle of decomposition from more difficult to elementary using only vertical links of hierarchy of construction elements of the reactor was used. This decomposition of construction elements of the ITER reactor begins with the main complexes of the ITER reactor and completes on the one-piece details or standard elements. Elements of structural functional model were coded in decimal system, therefore, so that each element of a design has the unique code defining its full identification in hierarchical structure of the reactor model (belonging to a certain complex or system, the hierarchy level, functional and constructive mission). Such way of identification allows to add to the developed hierarchical structure all processes of implementation of the project, including research and development, manufacture and assembly. However, in process of development of hardware and software possibilities of 3D-design tools, opportunities of databases, technologies of remote access, information technologies of administration systems and management of manufacturing processes, formations of concepts of electronic life cycle of products and a common information space extended ideas of opportunities of the developed concept.

Now the representation allowing to join of all activity connected with implementation of large scientific projects and to present it in the form of the interconnected systems of databases, three-dimensional models of object, temporary schedules of realization of tasks and representation of the reached results into integrated environment is created. Implementation of the developed integrated concept allow in future to increase essentially efficiency of invested funds due to minimization of losses from errors of design, the wrong administrative decisions, to optimize costs of research and development and production, to optimize technologies of assembly and commissioning, and also most seriously to reduce probabilities of realization of various types of risks.

Eligible for student paper award?:

No

M.OP3: Next Step Devices, DEMO, Power Plants / 437

Initial concept for the plasma diagnostic and control system for the European DEMO tokamak reactor

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The development of the plasma diagnostic and control (D&C) system for a future tokamak demonstration fusion reactor (DEMO) is facing unprecedented challenges. The DEMO D&C system has to operate with very high reliability, since any loss of plasma control may result in machine damage. On the same time, high accuracy of the D&C system is needed in order to allow for plasma operation near operational limits, where the fusion power is maximized. The implementation and performance of diagnostic components is however limited by space restrictions (optimization of the tritium breeding rate; integrity of the first wall and divertor against loads), and by adverse effects acting on the front-end components (neutron and gamma radiation, heat loads, erosion and deposition). Finally, the capabilities of the available actuators (poloidal field coils, external heating and fueling) are limited as well.

As part of the European DEMO conceptual design studies, the development of the D&C system has recently been launched ¹. A preliminary suite of candidate diagnostics for DEMO have been selected, aiming to cover all the main plasma control quantities with some redundancy, and choosing types and locations for diagnostic front-end components such that long-term durability with minimum need for maintenance can be expected under typical loads (e.g. neutron radiation, particle fluxes and fluencies). Specifically, only robust metallic or ceramics diagnostic components shall be used inside the vacuum vessel, while the more sensitive components shall be located at more remote (protected) positions. This initial plasma diagnostic suite comprises microwave diagnostics, magnetic coil based and Hall sensors, passive spectroscopy and radiation power measurements, divertor thermo-current measurement, infrared interferometry/polarimetry and neutron-gamma measurements. In the first R&D phase the possibilities and conditions for integration of diagnostic sightlines and front-end components into the machine have been investigated, and an understanding of the required number of channels and components has been obtained.

The conditions for controllability of the DEMO plasma are being analyzed by numerical simulations. To this purpose, the transport modelling code ASTRA, coupled to a radiation module, has been connected to the Simulink simulation framework, and the performance properties of diagnostics and actuators are being added [2]. Similarly, the existing CREATE equilibrium code is being amended to include limitations of diagnostics and actuators, such that the controllability e.g. of fast VDEs can be simulated under DEMO relevant settings. Furthermore, predictive control oriented models such as RAPTOR are being further developed as an alternative approach to analyze the controllability of DEMO. The common goal is to arrive at numerical simulations which closely mimic the control of the DEMO plasma such that the controllability based on the available diagnostics and actuators can be demonstrated quantitatively. Based on these R&D results, an initial version of the DEMO diagnostic and control concept has been elaborated and will be presented in this paper.

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Eligible for student paper award?:

No

M.OP1: Plasma Operation and Control / 438

New control ability on EAST PCS for steady-state operation

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EAST (Experimental Advanced Superconducting Tokamak), a toroidal device with a D-shaped poloidal cross section, aims at high confinement and steady-state operation with plasma current up to 1 MA and pulse length to 1000 s. To accomplish EAST physical targets, the plasma control system PCS, adapted from DIII-D PCS 1 and deployed on EAST in 2005, keeps in continuous development. Some new control abilities for steady-state operation has been achieved. One is the long pulse data acquisition/archiving using data segment technology of Mdsplus. The acquired raw data and calculated result of PCS can be read or analyzed by physical operators in real time, which will provide the possibility to adjust the control scenario during the plasma discharge [2]. Another is the loop voltage feedback controlled to realize the non-inductive operation. In 2016 EAST campaign, loop voltage is well controlled using low hybrid wave (LHW). Besides, another two control algorithms are implemented to reduce the divertor heat flux. One is radiation power control, which is successfully feedback controlled by using divertor inert gas puff and mid-plane supersonic molecular beam injection (SMBI). The other is quasi-snowflake (QSF) shape control using PEFIT/ISOFLUX, which shows significant heat load reduction to divertor target [3] according to the modeling and experiment result. In this paper, the strategy and implementation detail will be introduced. The steady-state ELM-free high confinement QSF discharge has been achieved with the pulse length up to 20s, about 450 times the energy confinement time. The present EAST PCS has become a huge system capable of long pulse, high performance advanced plasma control operation, which is ready to demonstrate ITER-like control contents.

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Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 81 / 439

Innovative H&CD designs and the impact of their configurations on the performance of the EU DEMO fusion power plant reactor

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Improvements of Heating & Current Drive (H&CD) systems are being investigated for a demonstration fusion power plant DEMO to deliver net electricity for the grid around 2050. Compared to ITER, which has to show the generation of 500 MW thermal power, the target of DEMO is the successful production of 300 to 500 MW electrical power to the grid and to aim for a self-sufficient Tritium fuel cycle [2]. Three H&CD systems are under development for DEMO in Europe, the Electron Cyclotron (EC) System, the Neutral Beam Injection (NBI) System and the Ion Cyclotron (IC) System.

Based on present studies [3] for plasma ramp-up, ramp-down and flat top phases, to be further validated in more detailed simulations, the assumed total launched power needed from the H&CD system in DEMO is in the range of 50-100 MW, to be provided for plasma heating and control.

Among other topics, the paper describes the new designs and R&D status of H&CD systems considered for their deployment in DEMO in Europe and the impact of the H&CD configurations on their performances based on those areas described in the European fusion electricity roadmap [4] for the integrated design and system development. These configurations encompass the operation of NBI with reduced Caesium consumption, with modular ion sources and improved injector wall-plug efficiencies, EC system with increased gyrotron frequencies above ITER ones, as well as multi-purpose and frequency step-tunable radio frequency sources, and related strategies for the fabrication of large Brewster angle diamond windows, together with EC launcher designs compatible with a fusion power plant environment, avoiding front-steering by using different launcher concepts.

The project also elaborates on new solutions to further increase the wall-plug efficiencies of H&CD systems based on more advanced concepts, with the target to reduce the recirculating power fraction in future fusion power plants. Different studies under investigation will be discussed such as, for NBI, the photo-neutralization and, for EC, new concepts for gyrotron multi-stage depressed collector.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission. With special thanks to whole WPHCD team for the durable, constructive and fruitful collaboration.

1 Federici, G., et al., 'Overview of EU DEMO design and R&D activities', Fus. Eng. Design, vol. 89, pp. 882-889, 2014

[2] Wenninger, R., et al., 'The physics and technology basis entering European system code studies for DEMO', Nucl. Fusion 57, 016011, 2017

[3] Vincenzi, P., et al, 'EU DEMO transient phases: main constraints and heating mix studies for ramp-up and ramp-down', SOFT 2016, accepted for publication in Fus. Eng. Design

[4] Romanelli, F., et al., 'Fusion electricity: a roadmap to the realisation of fusion energy', EFDA, Nov. 2012

Eligible for student paper award?:

No

R.OP4: Stellarators / 440**Preparation and commissioning for the LHD deuterium experiment****Author:** Masaki Osakabe¹**Co-authors:** Gen Motojima²; Hitoshi Miyake¹; Kunihiro Ogawa³; Kenichi Nagaoka⁴; Mamoru Sato⁴; Hiromi Hayashi⁴; Takuya Saze⁴; Kiyohiko Nishimura⁴; Sadayoshi Murakami⁵; Hiroshi Yamada⁴; Mitsutaka Isobe⁶; Kiyohiko Mukai⁴; Akihiro Shimizu⁴; Takanori Murase⁴; Yasuhiko Takeiri¹; Yasuo Yoshimura⁴; Naoyuki Suzuki⁴; Hiroshi Hayashi⁴; Masashi Kasaki; Tomohiro Morisaki⁷; Masahiro Tanaka⁸; Hiromi Hayashi⁴¹ *National Institute for Fusion Science, National Institutes of Natural Sciences*² *National Institutes for Fusion Sciences*³ *National Institute for Fusion Science*⁴ *NIFS*⁵ *Kyoto University*⁶ *National Institute for Fusion Science, National Institutes of Natural Sciences, SOKENDAI(The Graduate University for Advanced Studies)*⁷ *ational Institute for Fusion Science, National Institutes of Natural Sciences*⁸ *National Institutes for Fusion Science, National Institutes of Natural Sciences***Corresponding Authors:** kogawa@nifs.ac.jp, isobe@nifs.ac.jp, morisaki@nifs.ac.jp, osa@nifs.ac.jp, motojima.gen@lhd.nifs.ac.jp, miyake@nifs.ac.jp, mtanaka@nifs.ac.jp, takeiri@nifs.ac.jp

The deuterium experiment starts from March 2017 on the Large Helical Device (LHD). The deuterium experiment is a part of LHD performance upgrade project, including the upgrade of heating devices, the installation of closed helical diverter and the upgrade for diagnostic systems. The objectives of the deuterium experiment are:

1. to realize high-performance plasmas in helical systems by confinement improvement and by upgraded facilities;
2. to study the isotope effects on plasma confinement in toroidal plasma devices;
3. to demonstrate the confinement capability of Energetic Particles (EPs) in helical systems; and
4. to extend the research on the plasma-material interactions with long time scale using the benefits of the LHD's steady state operation ability.

The preparation for the deuterium experiments has been started since March 2013 after the conclusion of the agreement for the environmental conservation and the LHD deuterium experiment between NIFS and the local government bodies. As a preparation for the experiment, the tritium removal system was installed based on this agreement. The removal system removes all of Hydrogen isotope gases by converting them to water using catalyst from the exhaust of vacuum pumping system for LHD and all of its peripheral devices. The system can be utilized for the detailed analysis of mass balance of Hydrogen isotopes in LHD using the tritium produced in the deuterium plasmas as a tracer. This will provide useful information for future fusion reactor in evaluating the fuel mass balance. The Integrated Radiation Monitoring System (IRMS) was also installed as a part of central interlock system of the LHD. The IRMS monitors all of the signals from environmental radiation monitoring detectors as well as the signals from neutron detectors in the torus hall. It also monitors the status of doors at the boundary of radiation controlled areas and the personnel access to the areas through the access control system.

The two of Positive-ion based Neutral Beam Injectors (P-NBIs) were upgraded to improve their injection power from 6MW to 9MW by increasing their injection energies. The two of micro wave

launchers for Electron Cyclotron Heating (ECH) were moved from the upper port to the outer port close to the mid plane to increase the efficiency of ECH.

Three sets of neutron detectors, which consist of one fission chamber and one ^3He (or ^{10}B) proportional chamber, were installed as neutron flux monitors for LHD and the absolute calibration experiment for the detectors was performed using a ^{252}Cf neutron source in November 2016 as a part of commissioning for the LHD deuterium experiment.

In the presentation, the preparation and the commission for the LHD deuterium experiment will be discussed.

Eligible for student paper award?:

No

R.PLN: Plenary R / 442

Engineering Challenges in W7-X and preparations for the second operation phase

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In 2015 the optimized stellarator Wendelstein 7-X stellarator started with operation. The main objective of W7-X is the demonstration of the integrated reactor potential of the optimized stellarator line. An important element of this mission is the achievement of high heating-power and high confinement in steady-state operation. The approach to this mission is following three steps. First plasmas were produced in a limiter configuration (OP 1.1), then a test divertor unit is being installed (TDU) for the next campaign, OP 1.2, before the full steady state capability will be achieved implementing active cooling of all in-vessel components and a steady state high heat flux divertor.

In 2014/15, after the closure of the outer vessel, the commissioning of the W7-X device started. After the evacuation of the cryostat vessel and the plasma vessel, and checks of the mechanical stability of the vessels, the leak-search and cleaning of the 2000 m cryo-piping was started. In spring 2015 the magnetic coil set together with the support system was cooled down to 4 K. In the next step, the superconducting magnet system was loaded with currents for the first time. After integral commissioning of the magnet system, magnetic flux surfaces were confirmed using an electron beam.

In December 2015, the first helium plasma was generated using ECRH, in February 2016 the working gas was switched to hydrogen. The first operation phase (OP 1.1) was successfully finished in March 2016. At the end of OP 1.1 the discharge duration was close to 6 seconds, the limit for the heating energy was increased to 4 MJ and electron temperatures of ~ 10 keV were achieved. Due to the low densities in the range of 10^{19} cm⁻³ and the pure electron heating by ECRH, the ion temperatures reached only 2 keV.

At present, W7-X is undergoing the next completion phase, including the installation of the test divertor unit, the installation of the carbon tiles on the inner plasma vessel wall, an upgrade of existing diagnostics and the installation of new diagnostics.

This talk will discuss the engineering challenges in the preparation of OP 1.1, the first results from OP 1.1 and the further completion of W7-X and the preparation of OP 1.2.

Eligible for student paper award?:

No

M.OP1: Plasma Operation and Control / 443**Plasma control for EAST long pulse non-inductive H-mode operation in a quasi-snowflake shape****Author:** Bingjia Xiao¹**Co-authors:** Y.H. Wang¹; X.Z. Gong¹; G. Calabro²; F. Crisanti²; R. Albanese³; J.G. Li¹; Zhengping Luo; R. Ambrosino⁴; Q.P. Yuan¹; K. Wu; Y. Guo¹; G. De Tommasi⁵; D.A. Humphreys⁶¹ *Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, 230031, China*² *ENEA Unità Tecnica Fusione, C.R. Frascati, Via E. Fermi 45, 00044 Frascati, Roma, Italy*³ *ENEA/CREATE, Università degli Studi di Napoli Federico II, Italy*⁴ *ENEA-CREATE, Università di Napoli "Parthenope", Naples, Italy*⁵ *CREATE, Università di Napoli Federico II, Università di Cassino and Università di Napoli Parthenope, Via Claudio 19, 80125, Napoli, Italy*⁶ *General Atomics, San Diego, USA***Corresponding Author:** bjxiao@ipp.ac.cn

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

1 G. Calabro, et al, Nucl. Fusion 55 (2015) 083005;

Eligible for student paper award?:

No

T.POS: Poster Session T - Board: 34 / 445**Characterization of Low Energy Plasmas in the device PG-QRO-1****Author:** Gonzalo Ramos¹**Co-authors:** Martin Nieto-Perez¹; J. Julio E. Herrera-Velazquez²; Eduardo Andrade²; Oscar De Lucio²¹ *Instituto Politecnico Nacional*² *Universidad Nacional Autonoma de Mexico*

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The Plasma Gun PG-QRO-1 is a Coaxial electrode plasma discharge device with Mather type geometry. This geometry has been used in the past to develop Plasma Foci, which produce a large spectrum in energy, with energies ranging up to tens of keV. The study of the interaction of magnetized plasmas with candidate materials for fusion reactors, is a main topic in fusion research. The PG-QRO-1 device has been tailored to produce plasmas with relevant densities but limiting the high energy spectrum in order to use it for plasma-wall-interaction studies. We present here the study of plasmas of low energy produced with this device. The energy profile of the plasma is determined from the depth profile of samples of different materials exposed to deuterium discharge. The deuterium retention profiles in the materials are very shallow with penetration depths of the order of tens of nm.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 84 / 446

Anisotropic neutron emission spectrum and its utilization for verification of nuclear elastic scattering effect in proton-beam-injected deuterium plasmas

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It is well known that the nuclear elastic scattering (NES) contributes to the slowing-down of suprathermal ions in thermonuclear plasmas [1]. So far several calculations have predicted that ion heating by energetic ions, i.e. transferred power from energetic to bulk ions, is enhanced due to NES [2,3]. NES can also modify the fusion reaction rates [4]. It is important to experimentally ascertain the phenomena concerning plasma-heating process and validate the numerical simulations for fusion plasma operation and control. An observation scenario of knock-on tail due to NES by looking at the change of the γ -ray emission rate from $6\text{Li}+d$ reaction in a proton-beam-injected deuterium plasma has been proposed [5] and the experiment is just planning on Large Helical Device (LHD) at NIFS. When knock-on tail is created in deuteron velocity distribution function due to NES, the neutron emission spectrum by $D(d,n)^3\text{He}$ reaction is also modified from Gaussian distribution function with enhancement of the neutron emission rate itself from the value for Maxwellian plasma.

In this paper, the modification of the neutron emission rate and the double differential emission spectrum that will appear at the same time in the previously proposed experiment [5] are examined considering spatial ion behaviors in magnetic configuration [6]. We newly propose an observation scenario of knock-on tail using anisotropic neutron emission spectrum in proton-beam injected deuterium plasmas. It is shown that the modification of the emission spectrum is significant and non-Gaussian component appears approximately more than $\sim 1/50$ orders compared with the Gaussian peak. We also discuss a possible scenario for knock-on tail observation using both anisotropic, i.e., non-Gaussian, neutrons and γ -rays at the same time.

[1] J. J. Devaney and M.L.Stein, Nucl. Sci. Eng., 46, 323 (1971).

[2] J. Galambos, et al., Nucl. Fusion, 24, 739 (1984).

[3] Y. Nakao et al., Nucl. Fusion, 21, 979 (1981)

[4] H. Matsuura, Y. Nakao, Phys. Plasmas, 14, 054504 (2007).

[5] H. Matsuura, et al., Plasma Fusion Res., 2, S1078 (2007).

[6] H. Matsuura, et al., Plasma Fusion Res., 11, 1403105 (2016).

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 67 / 447

VARID: Virtual and Augmented Reality Integrated Development Facility for Research in Remote Handling and Maintenance of Tokamaks

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An efficient remote handling (RH) and maintenance scheme is the immediate requirement to ensure the maximum availability of the tokamak devices for the plasma operations. Virtual and augmented reality provides resourceful data to the RH operators for achieving the accurate control over the RH equipment and helps in time optimization of the RH operations.

The VARID facility established at IPR, India aims at research and development in the various areas of virtual and augmented reality that may prove extremely beneficial for the RH operators to have precise control of the RH operations without being physically present. The major research areas include computer vision, position tracking, 3d visualization, VR based real-time monitoring, haptic force feedback control system, kinematics & physics modelling etc. The facility also aims to act as a training facility for RH engineers to advance in the field of remote handling and robotics for future needs. The paper presents the design, layout and system configuration of the VARID facility. The initial results and achievements from the prototype experiments and trials are also highlighted.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 53 / 448

Assessment of Cavitation Erosion Risk in the Liquid-Lithium Flow in IFMIF-DONES

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The configuration of the Early Neutron Source (ENS) is the IFMIF-DONES (DEMO Oriented Neutron Source) approach, based on an IFMIF-type neutron source. It aims providing an intense fusion-like neutron spectrum with the objective to qualify on an accelerated time scale structural materials to be used in the future DEMO fusion reactor. IFMIF-DONES is based on the interaction of single 40MeV 125mA deuteron beam impacting a flowing liquid lithium target to simulate DEMO like neutron flux spectrum for fusion material irradiation experiments. Herein, the behavior of the high-speed (up to 15m/s) free-surface liquid lithium impacted by the deuteron beam is one of the key functional issues. Since it is practically unfeasible to avoid geometrical discontinuities such as steps, obstacles or gaps in engineering designs, a realistic assessment in terms of magnitude and location of the potential cavitation in the lithium system components is needed.

The present work focuses on the numerical investigation of cavitation phenomena in the lithium

flow at IFMIF-DONES relevant operation conditions. The Lithium flow is affected by different geometrical discontinuities on the channel walls. This has been simulated by means of an unsteady Reynolds –averaged Navier-Stokes (URANS) method. Calculations reproduce different cavitation processes depending on the kind of the wall surface discontinuity. In case of the flow over the lateral gaps, in the channel lithium gaseous phase generated within the gap remains stable and does not collapse. Simulations of the lithium flow over the backward-step show the generation of the gaseous lithium phase within the flow separation area and formation of a stable sheet cavity on the wall surface. The subsequent breakup of the sheet cavity in the flow reattachment region is accompanied by generation and collapse of unstable vapor structures downstream. The development and collapse of the vapor structures can induce cavitation erosion of the wall surface. The risk of cavitation induced erosion on the wall surface is assessed using a function based on the mean value of the time derivative of the local pressure.

The applied method provided the efficient identification of cavitation areas with high erosion risk in the Lithium flow systems. The knowledge obtained from the analysis result is used for the optimization of the Lithium flow conditions in the new design of the Lithium target and quench tank systems.

Acknowledgments

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

Eligible for student paper award?:

No

T.OP3: Project Management and Systems Engineering / 449

Project co-ordination challenges during W7-X completion

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The end of the first operation phase of Wendelstein 7-X, OP1.1, in March 2016 marked the start of an ambitious device completion campaign to install and commission a set of systems and components able to extract and support steady state operation from 2020 when OP2 is planned to start. Challenging assignments need to be completed such as the fabrication and installation of the high heat flux divertor and its cooling water supply system as well as the fabrication and installation of the plasma vessel cryo pumps and their cryo supply transfer system and extension. To manage the timely execution of these work packages the project “W7-X Completion”(W7-X/C) has been set up replacing the previous construction project. Furthermore, W7-X/C has to take into account and to shape the new role of the scientific and technical divisions as contributors of integrable, operational and safe components. The divisions will design, fabricate and partially enhance three auxiliary heating systems and around 50 diagnostic systems requiring intensive co-ordination and prioritization amid the natural resource split between technical project work and scientific exploitation of W7-X. Using the best part of the monitoring experience and where possible the established procedures, the role of Project Coordination (PC) has been tailored to suit the new requirements, away from mostly routine schedule and contract monitoring toward critical interface and priority management. The condensed time window until OP2 also calls for systematic early warning checks on both, time and finance schedules. Complementary to this, the new PC task of coordinating the revamped, more stringent design review process has become an effective tool to enforce systematic detailed work breakdown planning and review in the component supplying divisions.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 21 / 450

Observation scenario of knock-on-tail shape using Doppler-broadening

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The knock-on tail formed by nuclear elastic scattering (NES)¹ due to high energy particles gives various effects to fusion plasma. So far, observation experiment of knock-on tail was only conducted by measuring knock-on tail due to NES caused by alpha-particles using deuterium-tritium plasma at JET [2]. In this experiment, however, quantitative estimation of NES effect is insufficient. We proposed observation method of knock-on tail using gamma-ray-generating and neutron-generating nuclear reaction in a proton-beam-injected deuterium plasma at LHD. The gamma-rays and neutrons due to ${}^6\text{Li}(d,n){}^7\text{Be}$, ${}^6\text{Li}(d,p){}^7\text{Li}$, and $\text{D}(d,n){}^3\text{He}$ reactions will be utilized for the observation. However, it is difficult to identify the formed knock-on tail by using this method under high-temperature plasma conditions where the gamma-ray and neutron generation rates increase. In order to solve this problem, we newly propose the observation method using Doppler effects for the gamma-ray-generating reactions, ${}^6\text{Li}(d,n){}^7\text{Be}$, ${}^6\text{Li}(d,p){}^7\text{Li}$ reactions. It is also expected that the possibility to evaluate the shape of knock-on tail on velocity distribution function. In this study, we showed validity of this method for the proposed experiment by simulation.

[1] J. J. Devaney and M. L. Stein, Nucl. Sci. Eng., 46, 323(1971).

[2] J. Källne, et al., Phys. Rev. Lett., 85, 1246(2000).

Eligible for student paper award?:

Yes

M.OP2: Materials I / 451

The experimental investigation of wetting property for liquid lead lithium alloy with breeder blanket materials

Authors: Wei Lu¹; Weihua Wang¹; Haiyan Jiang²; Guizhong Zuo³; Baoguo Pan¹; Wei Xu³; Delin Chu¹; Jiansheng Hu³

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The dual-cooled lead lithium (PbLi) blanket is considered as one of the main options for the Chinese DEMO reactor. The liquid PbLi alloy is used as breeder material and coolant. The Reduced Activation Ferritic/Martensitic (RAFM) steel and the silicon carbide fiber (SiCf) are selected as its structural material and functional material respectively. In the present experimental investigation, the special vacuum experimental device has been built, and the 'dispensed droplet' modification of the classic sessile droplet technique has been used to investigate the wetting property and inter-facial interactions for PbLi/RAFM steel, PbLi/SS316L steel, PbLi/SiC and PbLi/SiCf couples. The contact

angles were measured between the liquid PbLi and the various candidate materials under working temperature from 300 oC to 480 oC. The results could provide meaningful compatibility database of liquid PbLi alloy and valuable engineering design reference of candidate structural materials and functional material for future fusion blanket.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 54 / 452

MOLECULAR DYNAMICS STUDY ON EFFECT OF GBS MISORIENTATION ANGLE ON GBS HELIUM EMBRITTLEMENT IN BCC IRON

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In fusion application, helium embrittlement is a key inducement to deteriorate mechanical performance of structural steels. To elucidate the mechanisms of helium induced embrittlement of grain boundaries (GBs), molecular dynamics (MD) was used to simulate GB tensile under the effect of helium bubbles in bcc iron at atomic level. Stress-strain curves and snapshots of configuration during tensile were extracted to analyze the mechanisms of GBs embrittlement. The GBs with <100> tilt axis and different misorientation angle which ranges from 5.5° to 84.5° were investigated under different helium concentration. Effect of misorientation angle on GB embrittlement was highlighted. The results indicate that the effect of helium bubbles on strength of GBs have significant dependency on misorientation angle of GBs. The study provides productive guidelines for the structural steels fabrication from the view of GB engineering.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 7 / 453

Investigation of the contact resistance between the pebble beds and the box wall surface in the gas flow condition

Authors: Seong Dae Park¹; Dong Won Lee¹; Dong Jun Kim¹; Suk-Kwon Kim¹; Jae Sung Yoon¹; Hyoung Gon Jin¹; Eo Hwak Lee¹; Mu-Young Ahn²

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The contact resistance effect in the interface between pebble beds and the structure was studied. The lithium ceramics is used as breeder with the form of sphere-shaped pebbles for the extraction of the

tritium in some TBM candidates of ITER. It could act as a thermal resistance in the interface and affect the pebbles and structural material temperature. Some models related to the contact resistance of the pebble beds were studied with the experimental results. The most of the heat transfer experiments were carried out with the limited box filled with the pebble. There is no flow motion of gas in the box. The flow of the gas is essential for the extraction of the tritium. The effects of the gas flow was considered based on the current models. CFD code, ANSYS-CFX 17.0 was used to evaluate the thermal performance in the box structure with and without the gas flow.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 55 / 454

Manufacturing design assessment of the welded in-wall shield rib for ITER

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Between inner and outer shells of vacuum vessel, numerous in-wall shielding (IWS) blocks are installed to provide neutron shielding. Uniquely, the IWS ribs in sector 1 and 6 are manufactured by welding and its manufacturing design shall be secured for not only manufacturability but safety point of views.

For design of IWS ribs, complex loads for multiple directional electromagnetic force, coolant pressure and earthquake shall be considered simultaneously. Moreover, in-service inspection is impossible after commissioning ITER therefore proper design assessment procedure should be applied with reasonable conservatism.

In this paper, an approach for manufacturing design assessment of welded IWS ribs is proposed. On the basis of that approach, four representative models are screened by geometric factors and load magnitude. Under the most conservative loads among multiple loading conditions, structural safety of each representative models are verified with RCC-MR. The deviation requests for this assessment was approved to ITER Organization and ANB. This approach and results could be used as a reference for design of vacuum vessel components which is impossible for in-service inspection.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 45 / 455

Monitoring, Modeling, and Protecting Against Insulation Failures in the NSTX-U TF Outer Legs

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The purpose of the effort described here is to model, and monitor the insulation shear bonds between the 3 conductors of the TF outer legs. Mechanical failure of the insulation could be a precursor to an electrical failure that could damage the more difficult to repair TF inner leg central column. The shear stress in these bonded layers is proportional to the TF outer-leg out-of-plane (OOP) bending. Bending of the outer leg due to out-of-plane loads and is supported partly by shear in the bond between the three conductors that are bonded together to form the outer leg. Bending stress in the outer conductors will provide an indication of the integrity of the shear bond. If the three conductors act together, as a beam, the metal bending stress in the outer conductors is as analyzed in the global qualification calculations. If the bond fails, then the bending stress will increase. This can cause a failure due to fretting motion in the insulation or overstress in the copper conductors, or failures in the water cooling tubes.

As a part of the orderly planned increase in operational parameters to .8 T (as of Feb 2016 it was at .61T), . Ten new FISO gauges were purchased and installed in March 2016. They yielded useful data prior to the forced shut-down due to the failure of PF1aUpper. The analytic process used to split out the thermal, in-plane and out-of-plane bending strains is described and compared with measured results. Measured results provide reasonable benchmarks for the analysis. Ultimately, the main purpose of the instrumentation is to compare coil to coil behavior, to watch for consistency. If a coil starts to deviate from the others we can inspect for possible de-bonding between the three conductors that make up the outer leg. To properly monitor the full TF system, more channels are needed than the FISO system can provide. The planned Fiber-Bragg Grating (FBG) system is introduced in the paper.

Monitoring and evaluation of the TF outer leg strains also is related to computed quantities monitored and protected by the Digital Coil Protection System. Currently the bending strain is expected to be adequately represented by upper-outer leg global moment sums. This relationship will be discussed in the paper.

Eligible for student paper award?:

No

M.OA2: Divertors and High Heat Flux Components / 456

Experimental and numerical investigation on anti-fatigue and anti-thermal shock performance of the divertor first wall

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After years of exploration and development, research of magnetic confinement nuclear fusion is progressed into stage of experimental fusion reactor construction and test. As a key plasma-facing component, the anti-fatigue performance of first wall of fusion reactor receives widely concerns. Due to the fact of enduring both periodic loads of pulse operating mode and shock loads of transient events such as disruption, ELMs etc, the coupled fatigue responses of material and structure are in the state of very complex. It is significant and necessary to research the coupled mechanism of fatigue by both transient and periodic heat loads, which will be beneficial to develop the key and new technology of promoting anti-fatigue performance for the first wall of fusion reactors. With such motivations, a multi-purpose experimental platform integrated both high heat flux loading and heat

shock loading as well as mechanical force loading is established. And meanwhile, a relative complete finite element analysis method based on a full coupled thermal/structural heat transfer equation with consideration of elastic/plastic constitutive relation as well as multiple kinds of thermal physical effects such as melting, solidification, evaporation etc. is established. Based both experimental and numerical works, the thermal/mechanical response of first wall and its fatigue performance are investigated. It is concluded that the fatigue life time of first wall is decreasing nonlinearly with increase of heat loads magnitude, and the coupled periodic normal loads and shock loads induced by transient events will greatly reduce the fatigue life time of first wall. And several techniques to improve anti-fatigue and anti-thermal shock performance are explored with both experiments and numerical tests.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 22 / 457

Design of a dual-band IR imaging system for surface temperature measurements on the tungsten divertor in EAST

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In 2018, EAST will be operated with a full tungsten (W) divertor in both the upper and lower divertors. Tungsten is a shiny refractory metal; as such, its emissivity in the infrared (IR) range is low. In addition film formation on the tungsten alters the emissivity, which makes precise surface temperature measurements difficult for conventional single-band IR cameras. To resolve this problem, a dual-band IR imaging system has been planned to install into EAST, to more accurately measure the surface temperature on the W divertor. The dual-band IR system has the advantage of being mostly independent of surface emissivity; using pyrometric techniques, the surface temperature is calibrated by the ratio of signals in two bands. A commercial single-band mid-wavelength IR camera combined with a two-band IR adapter is designed with a field of view 5.5×2.2. The two-band IR adapter utilizes a dichroic beam splitter, which reflects 3.7–4.2μm wavelengths and transmits 4.3–4.8μm wavelength radiation, each with >90% efficiency and projects each IR channel image side-by-side on the camera's detector. The dual-band IR images system will be used to monitor the upper outer W divertor with an existing mirror, with a ~1mm spatial resolution. In addition, a mirror installed into Material and Plasma Evaluation System [2] is designed for the measurement of the surface temperature on the lower outer divertor.

1 A.G. McLean et al., Rev. Sci. Instrum. 83, 053706 (2014)

[2] F. Ding et al., Journal of Nuclear Materials 455 (2014) 710–716

*This work was supported by DoE Contracts: DE-AC05-00OR22725, and DE-AC02-09CH11466

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 105 / 458

Engineering Design Modules on CFETR Integration Design Platform

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China Fusion Engineering Test Reactor (CFETR) integration design platform, which is intended to provide a unified environment to integrate physical and engineering design for future reactor-level fusion device, is now under development. It includes a physical design platform and various engineering design modules, such as vacuum vessel, divertor, toroidal/poloidal field coil, blanket, thermal shield, and neutronics, together with a standard material and design criterion database. Based on the experience of engineering design, the workflow and data flow of each engineering module are determined. Then, the interfaces within and among modules are implemented and used to integrate different modules into a complete framework. The interference check among modules is adopted in order to maintain the self-consistency of device design. The details of modules will be present in this conference.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 46 / 459

Studies on DEMO Toroidal Field Circuit

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The European demonstration nuclear fusion power plant (DEMO) is under conceptual design phase within the EUROfusion consortium. The most conservative design options in terms of science and technological developments with respect to ITER: with a $Q=10$, an operation with pulses 2 hours long lasting and the production of a net electricity power of 500 MW.

The toroidal magnetic field at the plasma centre of 5.7 T is produced with 18 superconducting toroidal field (TF) coils supplied by a current in the order of 65 kA, similar to the operating current of ITER TF coils, for a total stored energy of 136 GJ. This large amount of energy, more than three times the energy stored in ITER TF coils, has to be quickly dissipated in case of quench by the intervention of a suitable Quench Protection system (QPS). The time constant for the discharge for DEMO TF coils is 27 s, which is about 2.5 times the equivalent discharge time constant for ITER.

The energy, the current and the discharge time constant define the voltage to be applied to the coils;

however, the peak voltage at the coil terminals in case of faults can be much higher, therefore studies addressed to estimate maximum stresses in different operating conditions and to evaluate the relative merit of different circuit topologies are very important to drive the design work, aimed at finding the best compromise between requirements for the coil insulation and cost and size of the protection system, busbars and current leads.

Fusion devices usually make use of earthing circuits to fix coils ground reference so that the voltage applied to the two terminals versus ground is half the total voltage across the coil. Both ITER and JT-60SA exploit this scheme, with different topologies, that have been considered to be used for DEMO TF coils.

For JT-60SA and ITER the QPC is composed of a dc circuit breaker (CB) which normally bypasses a discharge resistor (DR); in case of quench the CB is open and the current is transferred to the DR. In addition, a backup protection based on a circuit breaker actuated via explosive is also provided. For DEMO further topologies have been explored, with the discharge resistors connected in parallel to the coil; this connection leads to a couple of benefits: at the QPS intervention, each coil discharges on its own resistor independently, and an opening failure of one QPC CB does not reduce the total resistance of the circuit.

When QPS intervenes, the same voltage is applied to all the coils and it is equally divided between the two terminals, with all the topologies aforementioned thanks to the symmetry of the circuit. When a fault breaks such symmetry the voltage applied between the terminals and the terminal voltage to ground can be different for each coil. The peak voltages depend on the fault and on the topology adopted for the circuit.

The fault analysis for the different topologies considered for DEMO and the results discussion are reported in the paper.

Eligible for student paper award?:

No

T.OP1: Power Supply Systems / 460

The ITER Power Supplies: status and recommendations for the next tokamaks

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The ITER Electrical Power Supply Systems comprise the Electrical Power Distribution (EPD), the Coil Power Supplies (CPS) and the Heating and Current Drive Power Supply Systems (H&CD PS). They will be connected to the French 400 kV power transmission grid, which will supply about 120 MW steady-state power required by the standard auxiliary loads and 500 MW, 200 Mvar pulsed power for the plasma scenario and control.

Although the design concepts adopted for several components are based on well-established industrial technologies, the huge installed power capacity, which exceed 3 GVA, and the presence of several, large, one of kind components make the ITER Power Supplies quite exceptional both in term of size and complexity. The overall system design and the design, manufacturing and testing of each main component include challenges that have been progressively addressed by design iterations, design studies performed by specialized companies and R&D. Furthermore, impressive test facilities have been built around the world to experimentally perform the proof of concepts and qualify the most demanding components.

At moment, some components required for the post-first plasma operational campaigns are still at the conceptual design phase. At the same time, some high power components of the ITER 400 kV substation are already installed and are ready for their first energization. Moreover, the mass production of the EPD and CPS components is well advanced in order to start the on-site installation at the end of this year.

The first part of the paper summarizes the main technical data, key facts and figures and present status for the procurement and installation of all the ITER Electrical Power Supply Systems. The main justifications which have driven the choice of the design requirements and technologies adopted for the main components are also presented.

The second part of the paper is intended for the designers of the future large Tokamaks and reports the most important lesson learned during the design, manufacturing and testing of the components that have been built so far. Moreover, considering that technologies and devices for high power components are evolving by increasing the power capacity and reducing the unit costs, the paper also reviews new technologies and industrial products that are emerging and should be taken into account in the trade off studies for the power supplies of future large Tokamaks.

Eligible for student paper award?:

No

M.OP2: Materials I / 461

EFFECTS OF TEMPERATURE AND HE CONCENTRATION ON FORMATION AND GROWTH OF HE BUBBLE IN BCC IRON UNDER IRRADIATION

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Helium atoms, produced at high rates in steels in fusion environment, are inclined to be deeply trapped in small vacancy clusters and microstructural features due to its low solubility in metal. Eventually, the formation of He bubbles significantly degrades the mechanical properties of materials. Therefore, it is important to understand the nucleation of He bubbles in steels, both in the bulk and within microstructural features, especially under irradiation.

In this presentation, the irradiation cascade damage process was simulated by molecular dynamics (MD) methods to investigate the formation and growth of He bubble in BCC iron under irradiation in which the energy of PKA is up to 200 keV. The effects of temperature and He concentration were analyzed. The temperature ranges from 300 K to 800 K. He atoms are randomly inserted into the iron matrix, either in tetrahedral or octahedral positions, and the corresponding He concentration is from 1000 appm to 3000 appm. The results distinctly show that the number of He-V clusters increases with increasing the PKA energy and dislocation loops with different types and sizes are produced. The formation and growth of He bubble is obviously faster with higher temperature and larger He concentration, respectively. Furthermore, the size of He bubble is almost distributed as the Gaussian distribution.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 8 / 462

Effects of the J-TEXT TBM mock-up on the equilibrium magnetic field and error field

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A scaled mock-up of the China Helium-Cooled Ceramic Breeder Test Blanket Module (CN HCCB-TBM) was installed to the J-TEXT in order to study eddy current distribution, electromagnetic load and thermal load on the TBM during plasma disruption. J-TEXT TBM mock-up using reduced activation ferritic/martensitic (RAFM) steel as structural material. The measurement experiments investigated the effects on the equilibrium magnetic field and error field in the presence of J-TEXT TBM mock-up containing ferromagnetic material. The experiments have measured TF ripple along the toroidal direction and the difference of vertical field over the plasma region by installing PCB poloidal array of magnetic probes inside the J-TEXT vacuum vessel with the mock-up at different positions. The error field introduced by the mock-up was evaluated by comparing phase difference and the amplitude variation of the signal of poloidal and toroidal Mirnov magnetic probes array. The amplitude and spatial phase of the error field were measured by scanning the spatial phase of an externally exerted resonant magnetic perturbation and fitting the mode locking thresholds. The experiment results show that the TBM mock-up in J-TEXT will change the shape of equilibrium magnetic field in vacuum vessel significantly, especially in low field side in front of the mock-up. The magnitude of 2/1 component of the radial error field introduced by the mock-up is about 10^{-4} relative to the base toroidal field. On the other hand, the effects will produce only a few potentially troublesome problems on the normal discharge.

Eligible for student paper award?:

No

R.OP6: Safety, Operations, and Maintenance / 463

Proposed methodology for unplanned repair scenarios in ITER

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Extensive nuclear analysis has been performed for ITER fusion device in order to determine the shielding requirements to minimize radiation exposure to operational personnel and sensitive equipment. The radiation exposure evaluation has been focusing on the areas in which personnel access is anticipated for planned maintenance, such as in the port interspace for diagnostic systems. This contribution focuses on the regions where hands-on repair activities may only occur if any components fail to function due to unexpected accidents or faults, and where short personnel access can be more efficient than remote repair. All ITER superconducting magnet components are housed within the cryostat and no maintenance is needed during nuclear operation of ITER. The ITER toroidal field coil terminal box region has been taken as a typical example for which the radiological impact is assessed and a methodology is sought to develop realistic and feasible repair scenarios.

The assessment has been conducted in four stages: 1) roughly estimate the contact dose for materials which have received different irradiation times; 2) calculate the shutdown dose rate counting all decay gammas in a simple rectangular box shape; 3) obtain and compare the shutdown dose rate in more detailed terminal box models (simplified from CAD models); 4) analyse the shutdown dose rate in the global environment with all other components included to contribute to the shutdown dose rate.

If components failed due to unexpected accidents, repair would always be before the end of ITER life

time. The calculations show that the contact dose rate is 20 times different between ITER life-time irradiation and 2-year irradiation therefore thorough testing and commissioning of components in the early part of the ITER operational life, designed to improve long-term reliability is essential. In case of an accident driven repair, we may propose countermeasures against excessive doses exposure by carefully planning the repair steps, routes and duration of work.

Moreover, with the ITER construction progressing, the phase has been moving from design to procurement and manufacture. At this stage of the project, procured components may be found to be made of materials containing different content of impurities to those in the specifications. It is generally possible to reject materials with higher impurity content such as cobalt which would create a local increase of the shutdown dose rate, an important parameter to estimate the radiation exposure level to radiological workers, but at the cost of replacing the component and a schedule delay. The methodology proposed in this contribution would also accommodate such materials reducing risk of failure in nuclear operation by an unrepairable component which cannot be easily foreseen in the stage of design and construction.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 88 / 464

Deuterium retention in tungsten exposed to KSTAR plasmas

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Tungsten (W) is considered to be the most viable armor material for the plasma-facing components (PFC) of a fusion reactor [1]. The work under fusion plasma will lead to modification of W that would change, in turn, its erosion properties, subsequent redeposition on surface, and would influence gas inventory (tritium, T) in material. Hydrogen (H) in W easily diffuses deep into W bulk even from the redeposited layers to the W substrate [2], namely, bulk T retention in W is a major safety concern. Erosion, deposition and deuterium (D) retention were investigated by installing marker tiles exposed to EAST plasmas in our previous work [3]. KSTAR is a superconducting tokamak with first wall fully covered by graphite tiles, and has planned a major upgrade to W first wall (coatings and bulk W-PFC) [4]. In this paper, D retention in bulk tungsten applying marker and redeposited surface have also been discussed.

7 sets of tungsten samples were installed in the lower divertor region and center column at the high field side during the 2015 KSTAR campaign, in which 2 sets were mounted at central divertor where could be observed by divertor IR camera from the top. Each set includes 2 marker W samples for erosion measurement and 2 polycrystalline W samples for retention study. To make sure that the marker layers could be measured after a long term plasma exposure with a possible change of the surface, SIMNRA are used in advance to design the marker including the variety of element, the thickness of the marker layer and the surface roughness. Tungsten coatings were then deposited on graphite substrate by magnetron sputtering as marker layers with a thickness of 400 ± 10 nm. The roughness of the substrate is around $0.1 \mu\text{m}$. And 4 deposited tungsten samples with different porosities were exposed in 2016 KSTAR campaign by divertor manipulator.

Surface morphologies and the compositions were characterized by standard surface analysis techniques, including scanning electron microscopy (SEM) and X-ray photoelectron spectroscopy (XPS). Rutherford backscattering spectroscopy (RBS) were used to measure the change in depth of the markers and deduce the net amount of erosion/ possible deposition on the samples. And the retention profiles were obtained by Nuclear reaction analysis (NRA) and thermal desorption spectroscopy (TDS).

- 1 V. Philipps, J. Nucl. Mater. 415 (2011) S2
[2] T Tanabe, Phys. Scr. T159 (2014) 014044
[3] J. Wu, W.R. Wampler, et al., J. Nucl. Mater. 438 (2013) S841-S845
[4] S.-H. Hong, K. -M. Kim, et al., Fusion Sci. Tech. 68 (2015)35

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 56 / 465

Study on helium-induced hardening due to interaction between helium bubbles and edge dislocation by molecular dynamics simulation

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Neutron irradiation-induced defects will degrade the mechanical properties of future fusion reactor structural materials. The understanding of the mechanisms for the interaction between gliding edge dislocation and irradiation-induced defects, such as voids and helium bubbles, is of vital importance. In this presentations, the interaction between an edge dislocation and helium bubbles with different sizes and he contents in bcc-Fe was investigated by using molecular dynamics simulation. Effect of temperature also have been considered in the simulation. The results indicate that the helium-induced hardening strength increases with increasing the size of helium bubble and decrease with increasing system temperature, respectively. The release stress of dislocation weakly depends on He/V ratio at the relatively low ratios, whereas a further increase of the He/V ratio leads to loop-punching from over-pressurized helium bubble. Thus the interaction mechanism was obviously changed at high He/V ratios.

Eligible for student paper award?:

Yes

M.OP3: Next Step Devices, DEMO, Power Plants / 466

Building a Virtual Tokamak - Integrated Multi-Physics Modelling for Fusion Engineering

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The design of any tokamak reactor presents one of the greatest engineering challenges in the world today, in particular for DEMO-class machines (which we define here as tritium self-sufficient, net electricity producing devices). By its very nature, such an endeavour requires the coordination of a vast effort spanning many fields, bridging physics and engineering disciplines. Typically, this activity is guided by the 0-1D systems code, PROCESS, which performs an extremely fast, preliminary,

single-parameter optimisation of plant design parameters to meet a set of input constraints and requirements. This is done in a matter of seconds, and the design point generated forms the basis of all further design studies and analyses. These activities cover an extremely broad range of different areas (e.g. superconducting magnets, breeding blankets, remote maintenance, etc.) and typically last one to two years before meaningful results can be fed back to the systems codes and another design point can be generated.

This work presents UKAEA's approach to bridging the feedback gap between the ~1s 0-1D systems codes and the ~1-2 year discipline-specific design studies. We present case studies that illustrate the first steps towards the realisation of a UKAEA advanced parameterised engineering design tool enabling the rapid generation of optimised tokamak designs: a tool to build an in silico tokamak.

The design for a parametrically engineered tokamak concept is presented, with a focus on the automated design of the superconducting toroidal field (TF) and poloidal field (PF) coils. We demonstrate some of the early capabilities of the code, including the capability to parametrically design, analyse, and optimise the superconducting coil cage of a tokamak (to the first order). A comparison study of the numbers of TF and PF coils is presented, and the resulting stored energy, superconductor volume, and cold mass calculated for each configuration. The impacts of different design philosophies are also assessed, such as adopting a super-X divertor geometry, or different first wall and vacuum vessel shaping strategies. Engineering tools for assessing structural integrity are combined with tools which are traditionally physics-based, such as equilibrium generators and coil placement optimisation routines, to evolve a viable plant concept in the form of automatically generated 3D geometry. This geometry is then used as a basis to measure the designs' performance analytically using more detailed structural, thermal, and neutronics codes.

The advanced parameterised engineering design tool is aimed at supporting engineering decision making, rapidly delivering the necessary substantiated designs and performance data to the designer, so that they may better understand the far-reaching implications of their design choices on the performance of DEMO-class tokamaks and fusion power plants as a whole.

Keywords: DEMO, systems codes, TF/PF coils, plasma equilibria, super-X divertor

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 100 / 467

3MW Dual Output High Voltage Power Supply Operation: Results for Accuracy, Stability and Protection Test

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High temperatures inside tokamak for fusion research is achieved from auxiliary heating systems like neutral beam injectors (NBI), or RF heating devices, viz., ion cyclotron (IC), electron cyclotron and lower hybrid systems where High Voltage Power Supply (HVPS) is an essential requirement. HVPS based on pulse step modulation (PSM) topology has already demonstrated its ability for broadcast transmitters, accelerators using radio frequency (RF) source and neutral beam injectors. For multi MW ICRF source, cascaded chain of amplifier is a practical solution due to limiting level of power with available vacuum tubes. Each chain of amplifier has to provide 1.5MW power in frequency range of 35- 65 MHz for 3600 seconds. The system must be capable to operate both at matched and mismatched load condition (VSWR 2). A novel concept of tapping two outputs from single PSM based HVPS is attempted for the first time. A PSM based HVPS is developed with dual output to feed driver and end stages of a high power RF amplifier.

Developed dual output HVPS is capable of providing 14 - 18 kV, 250 kW to driver stage and 16-27 kV, 2800 kW to end stage of a RF amplifier chain. Present article covers the validation of dual output HVPS in integrated operation with RF Amplifier system. HVPS performance parameters viz. ripple, regulation and stability over extended duration of 3600 seconds are presented for various scenario of RF Amplifier operation. Implemented scheme for protection against over voltage and over current is discussed. Besides, short circuit test conducted at the output of HVPS is also presented with setup, demonstrating tight synchronization among both stages. Prescription of gauge, length for fuse wire followed to meet the essential energy limit qualifications

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 109 / 468

Personnel Safety at Magnetic Fusion Experiments

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A part of the current fusion mission is to demonstrate that fusion experiments and fusion power plants can be operated in a safe manner for the workers, the local population, and the environment. This paper describes some of the present issues in personnel safety in magnetic fusion environments, including the present-day personnel protection limits. The historical trends of these protection limits are used to speculate on future personnel protection limits as fusion research advances from experiments to power plants and the difficulty in meeting more restrictive protection limits. Ionizing radiation, magnetic field exposure, chemical exposure, and radiofrequency energy exposure are addressed.

note: 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.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 47 / 469

Modeling and Qualifying Operational and Cooldown Strains of the NSTX-U PF1a Coils

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Two sets of three coils are positioned near the upper and lower divertor in NSTX Upgrade. These are collectively called the inner PF coils and consist of PF1a,b,c upper and lower. These are used in strike point position control, advanced divertor configuration studies, and Coaxial Helicity Injection (CHI) experiments. The NSTX Upgrade Inner PF coils have low Lorentz force derived stresses during operation. PF1a upper failed during the initial NSTX-U run, and this paper investigates the stress

state in the original and proposed replacement PF1a coils. The inner PF coils are sized based on temperature limits for long (5 second) pulses. More significant stresses and insulation strains develop during the cooldown process for the fully heated coil than during the energized, operational state. The current PF1a design is a relatively tall thin coil cross section with 4 layers of 16 turns each. Coolant is fed into the outside layer and extracted from the inner layer. The long coolant path produces cooling that progresses as a wave along the length of the conductor. At one point in the process, the outer layer is at the coolant temperature of 12C and the inner three layers can be as high as 100C. For the original PF1a coil the thermal stress in the outer conductor is close to the yield of the copper that was used, and above the fatigue limit set for NSTX-U conductors. For the replacement coil, yielding of the outer layer is allowed, and the resulting deformations have been qualified in terms of the cyclic shake-down of the copper, and strains imposed on the insulation system. Issues associated with qualifying cooldown insulation strains in the PF1a coils are similar to those faced in the qualification of the NSTX-U OH coil. Strain controlled cyclic displacement tests were performed on the OH insulation system which uses the same interleaved Kapton –glass system as planned for the pf1a coil. The tests indicate ample cyclic strain absorbing capability for the insulation system. The qualification of the cooling strategy and all operational stresses for the original and replacement PF1a coils is presented.

Eligible for student paper award?:

No

R.OP6: Safety, Operations, and Maintenance / 470

FORENSIC ANALYSIS OF FAULTED NSTX-U INNER POLOIDAL FIELD COIL

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On July 22, 2016, the NSTX-U project team suspended plasma operations due to the inoperability of the PF1A Upper (PF1A-U) coil. Preliminary indications evidenced that the PF1A-U coil experienced a coolant blockage. An external coolant leak developed from the PF1A-U coil pack after the blockage was attempted to be cleared. A post-mortem physics analysis indicated that an undetected gradual deterioration of coil inductance preceded the coolant blockage in the weeks leading up to the operational suspension.

The subsequent investigation, documentation, and analysis relative to the failure of the PF1A-U are the subject of this paper. Initial non-destructive testing and examination was followed by extensive non-destructive radiography. The radiographic study confirmed the locations of four braze joints and identified five anomalies. Destructive examination was initiated by segmenting the coil pack with cutting planes in feature-free benign regions identified by the radiography results. Cuts were cautiously made in 0.03-inch depth increments through the coil pack at the identified cutting planes. Two of the three coil sections were removed from the mandrel at the conclusion of cutting. The third section, containing the five anomalies identified in the radiographic study, the coil leads, and two braze joints, was initially left on the mandrel in order to minimize compromising evidence. All three sections were subjected to visual borescope/videoscope inspection through the cooling path, vacuum testing, and electrical testing of every conductor segment. One conductor cooling path visually evidenced a void through the sidewall of the cooling path in a layer-to-layer region. The identified void in the conductor cooling path was not proximal to a braze or joggle. Electrical testing indicated that the voided conductor segment was a member of a group of 14 conductor segments that evidenced low-resistance connectivity. The four braze joints and two lead segments were subjected to 400 psi hydrostatic pressure testing followed by helium leak testing and evidenced that there were no detectable leaks in the tested cooling path segments.

The coil section containing the conductor void was removed from the mandrel and split along a

layer plane to provide visual access to the conductor void area. A region proximal to the center of the coil pack evidenced electrical pitting and molten conductor debris. Samples of the epoxy resin insulation system were extracted and analyzed with Dynamic Mechanical Analysis (DMA) and Differential Scanning Calorimetry (DSC) techniques. Epoxy resin insulation samples were also subjected to immersion testing per ASTM D570. Metallurgical samples were extracted from the coil pack conductors and subjected to hardness testing and grain structure analysis. The sum results and interpretation of these analyses will be presented in the paper.

*This work is supported by US DOE Contract No. DE-AC02-09CH11466

Eligible for student paper award?:

No

W.OP2: Heating and Current Drive / 472

Conceptual Design of a 2-Channel Steady-State ECH Launcher for KSTAR

Author: Robert Ellis¹

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KSTAR will add a new 2-channel steady-state Electron Cyclotron Heating (ECH) launcher to its existing pair of launchers, providing 4MW of steady-state ECH power for heating and current drive.

This launcher is designed specifically for steady-state operation. Advanced features, such as fast steering with real-time position feedback for stabilization of neoclassical tearing modes, are included. Additive manufacturing is used for several components.

In this paper, the conceptual design of the 2-channel steady-state launcher is presented. Evolution of the design, starting from the original KSTAR ECH launchers, is summarized. The effect of emerging technologies, such as additive manufacturing, on the engineering of critical launcher components, is discussed.

Prototypes of some launcher components, including the mirrors, are now being fabricated. These components are described, and the results of initial testing are discussed.

*This work is supported by US DOE Contract No. DE-AC02-09CH11466.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 23 / 473

An Overview of NSTX-U Diagnostics

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NSTX-U will, because of its low aspect ratio and unique capabilities, be a critical element in world-wide magnetic fusion energy research. It has 6MW of high harmonic fast wave heating at 30MHz, 10MW of neutral beam heating, lithium evaporation capability for wall conditioning, and coils for control of resistive wall modes.

An extensive set of diagnostics is planned for multiple purposes: to enable plasma operation, to obtain data in support of the project's role as an international user facility, and to support NSTX-U collaborators.

The existing, and planned, NSTX-U diagnostics are described in the presentation. Those required for operation or of interest to multiple science groups are described in detail, including expected performance and results from initial NSTX-U experiments. Summary descriptions of the remaining diagnostics are provided. Engineering considerations in the design of NSTX-U diagnostics, particularly those unique to this device, are discussed.

*This work is supported by US DOE Contract No. DE-AC02-09CH11466.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 89 / 475

ACTIVE RECYCLING CONTROL THROUGH LITHIUM INJECTION IN EAST

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The coating of tokamak walls with thin layers of lithium has been demonstrated to reduce plasma recycling from the plasma-facing surfaces, and to improve overall plasma performance [1]. These effects, including reduced divertor $D\alpha$ emission, the elimination of edge-localized modes, and increased energy confinement have been observed in multiple experiments when lithium coatings are applied before plasma discharges. However, this coating technology does not extrapolate to future long-pulse devices, since the lithium coatings will be passivated by the continual plasma flux onto the surface. In order to provide active conditioning capability, a new technology has been developed that is capable of injecting lithium powder into the scrape-off layer plasma during plasma discharges, where it quickly liquefies and turns into an aerosol [2]. The use of this "lithium dropper" is under study at the EAST tokamak, where the potential benefits of real-time wall conditioning lithium injection are being tested.

Here we present an analysis of the recycling characteristics during EAST experiments testing active lithium injection in order to assess recycling reduction and control. Lithium aerosol was injected from the top of the machine, with one system dropping lithium near the X-point and another into the low-field side divertor leg. Lithium coatings applied via evaporation were applied at the beginning of the run day, and active lithium injection was intended to refresh the lithium layer later in the day after the original coatings were largely consumed. The injection of lithium into the SOL was reduced divertor recycling, as evidenced by reduced $D\alpha$ emission with ion flux measured by probes relatively unchanged. This effect is strongest in the divertor leg nearest the lithium dropper, becoming more pronounced with higher lithium injection rates. Quantitative analysis of the recycling changes during is underway using the SOLPS edge plasma and neutral transport code [3,4]. The assessment of the impact of lithium injection on divertor recycling coefficient and details of the dropper technology, and possible near term upgrades, will be presented.

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This work supported by the US DOE Contracts DE-AC05-00OR22725, DE-AC02-09CH11466, DE-FC02-04ER54698, and by National Magnetic Confinement Fusion Science Program under Contract No. 2013GB114004 and the National Nature Science Foundation of China under Contracts No. 11625524 and No. 11321092.

Eligible for student paper award?:

No

T.PLN: Plenary T / 476

Plasma Instrumentation for Spaceflight Missions

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Plasma measurements are an important part of spaceflight missions that seek to understand the formation and evolution of our solar system. Instrumentation has been designed for a wide variety of environments and measurement goals. We have developed plasma instrumentation that will fly within 9 solar-radii of the surface of the sun on NASA's Solar Probe Plus Mission. At the other extreme, we developed an instrument to measure the tenuous solar wind around Pluto at the edge of our solar system for the New Horizons Mission. At Earth, the Magnetospheric Multiscale Mission employs four spacecraft flying in formation to study magnetic reconnection on a global scale making measurements at unprecedented rates. While at Jupiter, the Juno Mission makes an in-depth study of Jupiter's polar magnetosphere to measure the effect of the precipitating particles on Jupiter's ionospheric layers, to determine the composition and structure of the field-aligned currents, and to understand the mapping of these currents to the outer magnetosphere and other parts of the Jupiter system. The instrumentation developed for these measurements spans a broad range of energies

from a few 10's of eV up to 100's of MeV. A wide variety of techniques and sensor technologies are employed to make the measurements, sometimes requiring special shielding and coincidence techniques to reduce background from the harsh space environment.

Eligible for student paper award?:

No

R.OP2: PMI and Plasma Edge Physics / 477

Utilization of isotopically enriched tungsten tracer particles and outer-midplane collector probes for impurity transport studies in the far scrape-off layer of DIII-D *

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First-of-its-kind experiments using isotopically-enriched, W-coated divertor tiles coupled with mid-plane collector probes (CPs) have been performed on DIII-D to understand divertor impurity production and transport. Inductively Coupled Plasma Mass Spectroscopy (ICP-MS) results are presented characterizing the isotopic ratios of deposited W on the mid-plane CPs and give quantitative information on the transport of W from specific poloidal locations within the lower outer divertor region. The setup includes two toroidal tile arrays (5 cm wide) of W-coated, TZM inserts in the lower outer divertor with the remaining plasma facing materials (PFMs) being carbon tiles. The inner ring was coated in natural-W (with 26.5% W-182) and the outer ring was coated with 93% isotopically enriched W-182. The unique "isotopic fingerprints" for the W impurities released from each coating in a dominant C PFM environment enables their use as tracer particles to be collected and distinguished at other locations. A triplet set of replaceable graphite CPs each with collection regions on opposing faces oriented normal to the magnetic field and distinct parallel connection lengths were mounted at the outboard mid-plane and inserted at the distance of $R-R_{sep} \sim 6-8\text{cm}$. Initial motivation for the CP is similar to that of W ring experiments on ASDEX Upgrade 1 and determination of the probe sampling (parallel connection) length was understood in the context of previous theory development of large probes in scrape-off layers (SOL) [2].

Rutherford Backscatter (RBS) analysis of these CPs has provided areal densities of elemental W content along the length of each CP face at 5 mm increments and found peak deposited densities on the order of 5×10^{14} W atoms/cm² and all CPs collected above the minimum RBS resolving threshold of 10^{12} W atoms/cm². These resultant W deposition profiles were compared with DIVIMP modelling of the far-SOL to better understand impurity transport in the edge plasma. ICP-MS analysis of the CPs performed at 5 mm increments along the probe length has successfully identified the presence of the enriched W isotopes and yielded isotopic ratios of the deposited W. By using a two-source Stable Isotope Mixing Model (SIMM) [3], the amount of W from each of the divertor rings that contributed to the total W deposition on the CP has been determined and shown to vary with the given plasma conditions, particularly ELM amplitude as examined through divertor spectroscopy and CP deposition profiles. An extensive comparison

of deposited W profiles with strike point positioning, H-mode/L-mode, ELM frequency, and forward/reverse B_t is reviewed.

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* Work supported by US DOE under DE-SC0016318, DE-AC05-00OR22725, DE-FG02-07ER54917, DE-FC02-04ER54698, DE-AC04-94AL85000.

Eligible for student paper award?:

No

W.OA2: Divertors and PFCs: Liquid Metals / 478

Synergies in Liquid Metal Technology Development for Divertor Applications*

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The difficulties with solid divertors for power handling in future reactors have long been recognized. In response to this challenge, the US initiated research into liquid metals as an alternative under the Advanced Power Extraction (APEX) program.^[1] To study the behavior of liquid metals with a large free surface in the presence of tokamak fields, a fully-toroidal Liquid Lithium Limiter (LLL) was installed in the Current Drive Experiment-Upgrade (CDX-U) spherical tokamak.^[2] The LLL was subject to high local heating with an electron beam, and convection in the liquid lithium was observed to distribute the heat load.^[3] The explanation was found to be thermoelectric magnetohydrodynamics (TEMHD),^[4] and is the principle behind the Liquid Metal Infused Trench (LiMIT) system being implemented as a limiter on the EAST tokamak.^[5] Lithium as a divertor target was also explored in the National Spherical Torus Experiment (NSTX) device, both as a coating on graphite plasma-facing components and a fully-toroidal Liquid Lithium Divertor (LLD).^[6] These approaches have addressed issues of particle control with lithium coatings and the stability of a liquid lithium layer on the textured LLD substrate.^[7] An additional effect that needs to be considered is ablation from the lithium surface under high heat loads. If the divertor is constructed with a box-like geometry, it could enable the lithium vapor to reach a sufficiently high density to extract momentum and energy from divertor plasmas.^[8] A prototype consisting of “vapor boxes”(VBs) attached together is being used to see if the configuration can separate the high pressure of lithium vapor from an outer vacuum chamber. For simplicity, the VBs are a series of cylinders that can be individually heated. If successful, the VBs can be tested with plasmas on the Magnum-PSI linear divertor simulator. Research in liquid metals to address specific engineering problems has thus uncovered new and unexpected phenomena, and broadened the options for their implementation in future divertors.

*Work supported by USDOE Contract DE-AC02-09CH11466.

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Eligible for student paper award?:

No

W.OA1: Materials II / 479

Tungsten-steel composites fabricated by roll bonding and ultrasonic welding for structural use in plasma-facing components

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Current fusion reactor designs often use a tungsten (W) to copper joint as part of the cooling structure in the plasma-facing components. Future fusion reactors may operate at temperatures above the operating window for copper. Therefore, robust joints between W and advanced steels are desired for fabricating plasma-facing components. A W-steel composite or functionally graded material is advantageous to minimize the stresses at the interface because of the thermal expansion coefficient mismatch between W and steel. Here, two methods of creating W-steel composites are examined: hot rolling and ultrasonic welding. Both methods utilize W foil because it has a ductile to brittle transition temperature below room temperature. Three initial thicknesses of W foil were utilized to fabricate the composites, 25 μm , 100 μm , and 250 μm . Before composite fabrication, each foil thickness has a different crystallographic texture and different grain size distribution. The differences in W foil properties resulted in different properties of the composites. The hot rolling method is a standard processing method and results in a significant intermetallic bond layer between the steel and the tungsten. The ultrasonic welding method is advantageous because it is a solid-state joining technique that reduces the thickness of intermetallic formed. However, ultrasonic welding of refractory metals presents other challenges such as a tendency of the W foils to shatter or delaminate during processing. The composites were analyzed with scanning electron microscopy and energy dispersive X-ray spectroscopy. Tensile and hardness tests were performed on the composites.

This work was supported by the Office of Fusion Energy Sciences, U.S. Department of Energy, under contract DE-AC05-00OR22725 with UT-Battelle, LLC.

Eligible for student paper award?:

No

R.OP5: Experimental Devices II / 480

Study of the impact of pre- and real-time deposition of lithium on plasma performance on NSTX

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Experiments in many machines have shown improvements on plasma confinement and edge stability with increasing levels of lithium (Li) conditioning. Different methods were used in these experiments to deposit lithium into the plasma, such as a laser-based injection of Li aerosol, injection by movable probe, use of a rotatable Li-coated limiter and use of a movable porous capillary Li limiter. During NSTX experiments with Li conditioning, three methods of Li injection were used. In this paper, the efficiency to improve plasma performance, in terms of the Li inventory, of two of these three methods of Li injection are compared. The first method evaporates Li over the lower divertor targets using a device that consists of a reservoir oven with an output duct that is inserted into a gap of the NSTX upper divertor –the “LITER”. Typical evaporation rates obtained with the LITER are in the range 1 to 40 mg/min. In this method, the Li thin-film is deposited on the divertor targets before the discharge where the amount of injected Li depends on the amount of time that LITER is in operation. The second method uses a device that injects a Li aerosol into the plasma during the discharge by simply dropping spherical Li powder in a controllable manner using a vibrating piezoelectric disk with a central aperture –the “Li dropper”. Controllable real-time injection of Li aerosol offers some advantages over the thin-film pre-deposition method, such as the real-time replacement of the Li thin-film removed from the divertor targets by the plasma-wall interaction. Typical evaporation rates obtained with the Li dropper are significantly larger than with the LITER and range between 1 to 120 mg/s. However, considering that the LITER evaporates Li during several minutes before the discharge, and that the Li dropper does not operate for more than 1-2 seconds, the total amount of Li injected with these two methods are comparable. The results show that the Li dropper is more efficient than the LITER, as it requires lower amounts of Li to cause the same performance improvements obtained with the LITER. However, the results using the Li dropper show that, above an evaporation rate of about 80 mg/s, no significant incremental change in plasma performance is observed. The effect of the Li evaporation rate on recycling and kinetic profiles for these two methods of Li injection will also be discussed and compared with those in discharges where the plasma facing components were conditioned by helium glow discharge cleaning and boronization. This research was supported by the General Atomics Postdoctoral Research Participation Program administered by ORAU and is part of the General Atomics Collaboration on Plasma Boundary Interfaces and Macroscopic Stability at NSTX-U. This work has been supported by the US Department of Energy, Office of Science, Office of Fusion Energy Science under DOE award DE-SC0012706.

Eligible for student paper award?:

No

R.OP2: PMI and Plasma Edge Physics / 483

Study of D retention and impurity emission properties of oxidized B4C coatings under deuterium irradiation in NSTX-U

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Conditioning and irradiation induced modifications on Plasma Facing Component (PFC) materials play a key role in the plasma performance in tokamak machines. Boronization is a conditioning technique widely used in carbon based machines due to its associated sputtering reduction and oxygen gettering properties. The National Spherical Tokamak Upgrade (NSTX-U) used boronization with d-TMB during the 2015-2016 experimental campaign as the main PFC preparation method. The Materials Analysis Particle Probe (MAPP), an innovative PFC chemical characterization facility, recently commissioned in NSTX-U, was used to investigate the evolution of boronized ATJ graphite when irradiated with energetic D ions in the Lower Divertor (LD) of NSTX-U. MAPP was used to

insert graphite samples into the outboard LD of the tokamak and retract them (with no exposure to atmosphere) into an analysis chamber for X-ray Photoelectron Spectroscopy (XPS) interrogation every twenty four hours. The results show that a thin boron carbide film is deposited onto the carbon PFCs following each boronization procedure. Furthermore, the XPS data indicate that the oxygen content of the samples increases with plasma exposure, going from 5% to almost 20% after tens to hundreds of plasma discharges. Moreover, comparisons of the duration of the plasma discharges pre and post boronization seem to correlate these incremental increases of oxygen content with the performance of the machine. In addition to the characterization of ATJ samples, MAPP also exposed a TZM (Ti, Zr and Mo alloy) sample during the campaign. Similar XPS analysis was carried out with this sample, and the results indicate no chemical interaction between the B coatings and the TZM sample. However, the boronization procedure covered the substrate beyond the probed depth of XPS. As was the case of the ATJ samples, oxygen plays a key role in the evolution of the chemistry of the TZM surface. TZM is a candidate material to replace graphite in the lower and upper divertors of NSTX-U, hence a detail investigation of its behavior under machine conditions is highly relevant. Such observations are only possible due to the improved time resolution and surface sensitivity (20 nm) of MAPP. The benefits associated with this diagnostic are twofold. Firstly, it can provide light on plasma-surface interactions and provide correlations with PFC surface conditions at the fundamental level. Secondly, its in vacuo surface analysis capability can be used to give an updated description of the PFC state to machine operators and physicists to guide their decisions on surface conditioning during the experimental campaign.

*Work supported by USDOE Contract DE-AC02-09CH11466.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 101 / 484

The Proposed Improvement for Neutral Beam Injection Power Supply System

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Since the construction of previous generations of fusion reactors, new technology has emerged that can enhance the performance of the high power, power supplies (HPPS) used in fusion energy. For example, wide bandgap (WBG) based switching devices have emerged with ratings suitable for some elements of power conversion. Furthermore, modular multilevel converters (MMC) are a topology that have gained in popularity and are maturing quickly. This research explores utilizing both of these technologies in the HPPS for the auxiliary heating systems of a tokamak reactor. The Experimental Advanced Superconducting Tokamak (EAST) is used as a case study for this work. Although the results are applicable to other tokamak based fusion energy systems.

WBG devices have a unique molecular structure that enables power devices to be created with lower on-resistance. Additionally, WBG components have faster switching speeds and that reduces switching losses. In addition to these advantages, WBG devices have the potential to realize higher breakdown voltages. This makes them an attractive candidate for implementation into alternative topologies for the HPPS, such as MMCs. This work extends that study presented in to include the entire power supply module (PSM) and considers the benefits of using WBG components in all of the auxiliary heating systems.

The current neutral beam injectors (NBI) require a 100 kV/100A HPPS. By utilizing an MMC topology over the existing method, several advantages can be realized. A single MMC can eliminate all but one of the transformers that are used in the existing NBI supply and it eliminates the rectifiers in the power supply modules (PSM). Additionally, it is possible for the MMC to act as a power supply and harmonic filter hybrid. The inclusion of WBG power devices into an MMC also provides opportunities to modify the existing submodule construction.

The final paper will analyze the impact of using WBG components in the existing auxiliary heating systems. It will examine how an MMC based rectifier can be used instead of the existing power supply topology. It present methods for using the MMC rectifier as a power supply and a harmonic filter. Finally, it presents the benefits achieved by combining all three of these technologies.

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 33 / 485

Engineering overview of the Fusion Research in Costa Rica: SCR-1 Stellarator and Spherical Tokamak MEDUSA-CR

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As of this day, two major magnetic fusion research projects are held at the Plasma Laboratory for Fusion Energy and Applications at Instituto Tecnológico de Costa Rica (Costa Rica Institute of Technology). The current status of both devices is summarized.

On June 29, 2016, the Stellarator of Costa Rica 1 (SCR-1) produced its first hydrogen plasma, becoming the first Stellarator of Latin America and one of the few operatives in the world. This fusion research device was fully designed, constructed and implemented in Costa Rica. SCR-1 is a 2-field small modular Stellarator ($R=0.247$ m, $\langle a \rangle=0.040$ m, $R/a=6.2$, plasma volume ≈ 0.0078 m³, 10 mm thickness aluminum torus shaped vacuum vessel) 1. Plasma is confined using a magnetic field of 43.8 mT generated by 12 modular coils with 6 turns each. The SCR-1 plasmas are heated by ECH second harmonic at 2.45 GHz with a plasma density cut-off value of 7.45×10^{16} m⁻³. Two magnetrons with a maximum output power of 2 kW and 3 kW are used. Currently, the SCR-1 is at a validation process with magnetic mapping tests, set to determine the quality of the magnetic field confinement, and the plasma shape. The different tests performed are, oscillating rod mechanism, fluorescent screen, and in house developed method combining the last two.

Furthermore, different vacuum tests have been performed on the toroidal vacuum vessel to optimize the pressure levels. Validation results for the modular coils, vacuum vessel, and an explanation of the construction process of each component are presented. Also, the design and implementation of the coils' electric current regulator, and the acquisition and control system are detailed.

The low aspect ratio spherical tokamak (ST) MEDUSA (Madison EDUcation Small Aspect ratio tokamak), donated by the University of Wisconsin-Madison, USA. is currently being re-commissioned at Instituto Tecnológico de Costa Rica. The major characteristics of this device are: plasma major radius $R_0 < 0.14$ m, plasma minor radius $a < 0.10$ m, plasma vertical elongation 1.2, toroidal field at the geometric center of the vessel $BT < 0.5$ T, plasma current $I_p < 40$ kA, $n_e(0) < 2 \times 10^{20}$ m⁻³, central electron temperature $T_e(0) < 140$ eV, discharge duration is < 3 ms, top and bottom rail limiters, natural divertor D-shaped ohmic plasmas). As part of the recommissioning process, several technical tasks are being performed, such as the re-design and implementation of the gas injection,

vacuum systems, and the re-design of the coils' electric current control system. Progress in some of these topics will be presented in this work.

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Eligible for student paper award?:

No

W.OA2: Divertors and PFCs: Liquid Metals / 486

Advancement of LiMIT and Associated Technologies

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Liquid metal plasma facing components (PFCs) provide several advantages over standard solid PFCs, as the constantly refreshing, self-healing surface reduces erosion and thermal stress, decreases edge recycling, reduces impurities, and enhances plasma performance. The Liquid Metal Infused Trench (LiMIT) system, pioneered at UIUC and tested in HT-7 and Magnum PSI, has demonstrated the feasibility of thermoelectric magnetohydrodynamic (TEMHD) driven flow of liquid lithium through a series of solid trenches. As the system moves closer to full-scale implementation in fusion devices, the team at UIUC is focusing on addressing issues of flow control and heat flux handling, as well as the potential for the use of novel materials in a LiMIT system. We present advances in the use of dictating wetting temperature and therefore flow regimes using a novel surface micro/nanostructuring process, and the potential improvements to system behavior these techniques can yield under high heat loads. We also investigate the potential of a tin-lithium (SnLi) eutectic as a viable alternative in the LiMIT system. Since the advent of liquid lithium PFC concepts, SnLi has been discussed as a possible perfect compromise between the stability of tin and the plasma performance benefits of lithium. With an effective process for creating bulk SnLi eutectic at UIUC, we have been able to test several fusion relevant characteristics of SnLi, and investigate its potential as an improved liquid metal PFC material.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 34 / 487

PHYSICS OF THE HIGH FIELD ULTRA LOW ASPECT RATIO TOKAMAK

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A medium-size High toroidal magnetic Field Ultra Low Aspect Ratio Tokamak (HF-ULART) is proposed. The major objective of this is to explore the highest beta limit possible under the maximum

toroidal field(TF) to have also high plasma pressure, using present day technology and achievements of tokamak fusion research.

This is the right pathway scenario to initiate studies for a potential ultra-compact pulsed neutron source(UCPNS) based on the spherical tokamak(ST) concept, which later, may lead to more steady-state neutron source or even to a fusion reactor, via realistic design scaling.

The major characteristics of this device are: plasma major radius $R_0=0.52\text{m}$, plasma minor radius $a=0.42\text{m}$, aspect ratio of $A=1.24$, plasma vertical elongation $k=2$, triangularity $\delta=0.9$, TF at the centre of the vessel of $B(R_0)<1\text{T}$, plasma current of $I_p<0.5\text{MA}$, central density of $n_e(0)=1e20\text{m}^{-3}$, central electron temperature of $T_e(0)=1\text{keV}$, and discharge duration of $t=100\text{ms}$.

The vessel is spherical, made of stainless-steel, and insulated from the natural diverted(ND) plasma by thin(few centimeters) tungsten(W) semi-spherical limiters. No internal poloidal field coils or solenoid is envisaged. This helps the compactness (relative close plasma-vessel fitting in order to capitalized of potential wall stabilization as envisaged in the RULART proposal¹) and to easy plasma/neutron shield via a thin(2mm) W bored rod covering the cooper central stack. This might be the only pre-cooled component by liquid nitrogen flow, thus maintaining the whole design simple and cheap.

The major source of initial heating is provided by I_p generated from RF in combination with Coaxial Helicity Injection(CHI) techniques, as both have been successfully demonstrated separately in STs: RF in LATE/QUEST and CHI in NSTX/Pegasus.

After a very high beta configuration is attained (potentially in H-mode as observed in Pegasus ohmic H-mode in natural divertor configuration using inboard gas fuelling), adiabatic compression(AC) technique is applied via raising $B(R_0)$ to higher values($<2\text{T}$) and possibly synchronised with the raise of I_p to 1MA, both for short period(few milliseconds) in a similar way was conducted in TUMAN-3 tokamak. At the peak of AC phase, single cryogenic pellet injection followed by neutral beam injection(NBI) heating are used for further raising $T_e(0)$ in a high density peaked profile target, leading to high neutron yield, similar to PEP(JET) or super-shot(TFTR) high performance discharges.

This HF-ULART can help the revival of the use of the AC technique in tokamaks, alongside the ST-40, a larger, less compact, and more complex device, currently under construction^[2]. In addition, studies in HF-ULART as a UCPNS help also to test the feasibility of similar compact neutron source via the spheromak concept with the AC technique^[3].

Preliminary equilibrium and stability simulations prior the use of the AC technique will be presented, and basic neutronic calculations, at the peak of this phase. Constraints of plasma power load in the vessel, plasma control and gas fuelling (inboard and outboard) systems, and central stack design, will be also discussed.

1C.Ribeiro, Proc. 26th Symposium on Fusion Eng.(SOFE), Austin, 2015.

[2]M.Gryaznevich et al., 18th International Spherical Torus Workshop(ISTW), Princeton University, 2015.

[3]P.Sieck et al., US-Japan Workshop on Compact-Torus, Irvine-CA, 2016.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 90 / 488

Dynamics and control of droplet splashing from tungsten divertor materials generated by ELM-like heat loads

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The ELM-like transient high-heat flux generates melt-layer formation of tungsten (W), melt motion and droplet ejection, leading to surface erosion of plasma facing components in large fusion devices such as ITER. This paper will present the experimental investigations of dynamics of W droplet splashing with including the stabilization effects of the magnetic field, which have been performed by using the magnetized coaxial plasma gun SPICA facility at NIFS.

In the experiment, we have demonstrated the melt layer erosion and splashing on two W target plates installed with a difference in level in a chamber. These phenomena were observed under the condition of ITER-ELM relevant heat loads of $2\sim 4$ MJ/m². The surface temperature measurement indicates that the W surface temperature increases rapidly up to the melting temperature of 3695 K within the plasma pulse duration of 0.12ms. The peak gun current is 200-300 kA at the charging voltage of 15-23 kV. The velocity of hydrogen plasma stream is 120-160 km/s. The electron density of plasma is 2×10^{21} m⁻³. The angle between the target surface and the plasma stream is set to 45 degree. There are traces of coagulation of melting W and bridging of gaps due to melt motion on the damaged area (30x40 mm). We identified droplets emitted from the W target surface by using a high speed camera. The droplets are ejected in the parallel direction to the plasma stream. The moving of droplets can be seen for $t=1.5$ ms after the plasma impact terminates at $t=0.12$ ms. The droplet continues to be ejected from the melt layer after the impact. The droplet speed is about 26 m/s at $t=0.3$ ms and then slows down to 13 m/s at $t=1$ ms. The JxB pinch force produces the droplet ejection of the melt layer due to the plasma pressure.

We have applied externally the magnetic field $B < 0.15$ T parallel to the direction of plasma stream. It has been found that the magnetic field could decrease the droplet speed and suppress completely the droplet splashing as B increases. Also, the suppression efficiency depends on the direction of the parallel magnetic field. One possible explanation is that the propagation of surface waves such as Kelvin-Helmholtz instability is damped by the imposed magnetic field parallel to the W-melt flow and the plasma stream near the surface, so resulting in suppression effects on the development of droplets.

Eligible for student paper award?:

No

M.PLN: Plenary M / 489

Overall Status of the ITER Project

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The ITER project, established by an international agreement among seven Members (China, the European Union, India, Japan, Korea, the Russian Federation and the United States of America), is a critical step in the development of fusion energy: its role is to confirm the feasibility of exploiting magnetic confinement fusion for the production of energy for peaceful purposes by providing an integrated demonstration of the physics and technology required for a fusion power plant. Rapid progress has been made over the past two years in the design, manufacturing, construction and R&D activities, and the facility is now taking shape at St Paul-lez-Durance in southern France.

Supported by impressive achievements in fusion technology R&D, manufacturing of ITER components is advancing in factories and laboratories around the world. The international collaboration formed around the production of superconducting magnets for the ITER tokamak has produced over 600 t of Nb₃Sn and 300 t of NbTi superconducting strand. 90% of the superconductors required for the ITER magnets are now complete, contributed by 6 out of the 7 ITER partners. Winding packs for the first 4 toroidal field coils have been produced and stacked in the EU and Japan, and central solenoid and poloidal field coil fabrication activities on the first-of-kind coils are underway in partners' factories in China, France, Russia and USA. Successful tests of prototype high temperature

superconducting leads for ITER magnet systems using Bi-Sr-Ca-Cu-O (2223) tapes have also been completed and series production of the current leads has been launched. Fabrication of the vacuum vessel is moving forward, with structures being manufactured under the responsibility of four contributing Domestic Agencies. Manufacturing of the thermal shield is also in progress, and the cryostat elements delivered to the ITER site by India are currently being assembled into large-scale sections of the cryostat (~29 m diameter × ~29 m height). Substantial elements of the power supply and cryogenic systems have also been delivered and several captive (water) drain tanks have been installed, the first equipment incorporated in the Tokamak Complex and the first steps in a multi-year on-site installation programme of tokamak and plant systems which is about to be launched.

ITER Management is continuing its efforts to strengthen project integration, streamline decision making and ensure the efficient use of project resources while accelerating construction activities. During 2015 and 2016 the ITER Organization and Domestic Agencies worked closely to redevelop the project baseline schedule, providing a realistic framework for the completion of construction while meeting the Members' budget constraints. The 'staged approach' strategy endorsed by the ITER Council in November 2016 has established a target for First Plasma of December 2025 as the earliest technically achievable date, with the transition to DT operation scheduled for December 2035.

The presentation will review the progress made in developing the advanced technologies required for ITER and in the manufacturing of major components, describe the status of construction of the ITER facility, discuss measures taken to establish a more effective project organization and summarize the revised baseline schedule.

Eligible for student paper award?:

No

R.OP3: Tritium Extraction and Control / 490

Validation of Tritium Self-Sufficiency of DEMO

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The authors have suggested that realistic Power Ascension Tests (PAT) of DEMO can produce its tritium to be needed in the series of tests by its own operation from initial DD discharge until reaching steady state full power burning with no external supply. It is generally understandable that closed tritium fuel plant can collect small amount of tritium to be produced by DD and DT reaction and breeding in the lithium containing blanket within reasonable dwell operation following each discharge to be anticipated in realistic operation scenario. Fuel system was described by a system dynamics model, and analyzed considering realistic PATs of DEMO, that will be mainly pulsed DD and low concentration DT. Primary fuel cycle is composed of plasma exhaust evacuation, isotope separation by cryogenic distillation, storage and blanket tritium recovery. Secondary systems is such as tritium recovery from water and solid waste, secondary confinement to capture permeated and leaked tritium are also considered. Tritium returning time constant varies depends on the function of the each components, and finally collected at the storage after the staged recovery from the components according to their tritium release time constant.

Although no actual PAT plan for fusion DEMO is available, previous PATs for new fission reactors and recent consideration of ITER operation plan can provide sufficient possible scenario of initial commissioning operation of DEMO. Typical PATs will require years of operation from zero power that would be DD discharge, with pulsed power output and long dwell time between them. Output power will gradually be increased in PATs to check the functions of reactor systems and components. Although components of the tritium fuel cycle in the primary loop can be anticipated based on the previous tokamak operation experience and Tritium System Test Assembly at LANL, breeding blanket would be the most unknown and critical components on this fuel self-sufficiency of the DEMO reactor. Possible validation experiments and methodology to measure actual tritium production capability with satisfactory accuracy will be presented.

Eligible for student paper award?:

No

R.OP3: Tritium Extraction and Control / 492

Liquid PbLi atomization in vacuum for tritium and heat recovery

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The authors proposed a concept of tritium and heat simultaneous recovery from lithium-lead (PbLi) in vacuum in 2016. Hot and tritium rich liquid PbLi is transformed into small size droplets (atomization) through nozzles in a vacuum recovery chamber. While falling, tritium is released by advection mass transport and recovered by a vacuum pump. Heat is transferred by radiation to the counter-flow secondary medium through a chamber wall. Both mechanisms can function in vacuum. The first hurdle of this concept is the PbLi atomization in vacuum. Conventional spray mechanism in air or in gas, is not applicable in vacuum. Hence different instability mechanism which occurs on a velocity inflection is applied. By a preliminary experiment using a water in room air, surface mean droplet diameter was shown to be as small as 0.17 mm at the velocity of 10 m/s. It obeyed approximately power of minus 0.5 on the flow velocity. Obtained mean diameter was small enough for higher than 90% tritium recovery efficiency within 1 second of vertical drop, by a previous study result. As for the heat recovery, design window was identified by the case study as the function of the temperature difference (ΔT) between droplet and surrounding wall, and the emissivity(ϵ). PbLi droplet temperature of 973K in and 823K out is expected to be possible with (ΔT , ϵ) of (100K, 0.5), (150K, 0.4) and (200K, 0.3). However, even a same material, many factors were reported to strongly affect the emissivity. An experimental setup using a low temperature melting alloy (GaInSn) in vacuum is under fabrication for further verification. The atomization in vacuum and the emissivity value of GaInSn spray will be verified before final experimental using PbLi.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 68 / 493

Design of the dimensional metrology and alignment scheme for the 1/32 CFETR VV Mock-up

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Abstract: The vacuum vessel (VV) of Chinese Fusion Engineering Testing Reactor (CFETR) is a D-shape, double-layer and toroidal structure, which have a high precision requirement, a large-size and too weight, the challenge can be foreseen in the VV manufacturing. As an important control means of the quality, the hyperboloid surface of VV should be inspected to ensure qualified manufacture such as heat form, weld and machine, however the inspection of the hyperboloid surface and assembly alignment are difficult issues. In this paper, a scheme of three-dimensional (3D) metrology and alignment was proposed, which were welding the special nests of fiducial on VV, using Laser

Tracker to measure the surface of VV, computing and fitting the measurement points to designed model, getting the best fit and the average errors by setting the weight of points and changing the calculation model, then calibrate and value the fiducials which can be using in after manufacture and assembly as the datum. The scheme of 3D metrology solve the difficult issues of the inspection of the hyperboloid surface and assembly alignment during in 1/32 CFETR VV Mock-up manufacturing which was completed by ASIPP.

Keyword: Inspection of the hyperboloid surface, Assembly alignment, Best fit, Laser Track.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 32 / 494

Improved thermal performance of an updated NSTX-U inner divertor

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An improved inboard divertor has been designed for NSTX-U with higher divertor heat flux capability, robustness to halo current strikes and improved bakeout temperatures. The ATJ graphite inboard divertor tiles of the original NSTX-U design may have been subject to radial forces from halo currents which were not effectively resisted by the t-slot tile clamping. The clamping force could have been increased to resist these forces, but the stress in the t-slot cut of the tiles would have left insufficient margin for the thermal stresses induced by the divertor heat flux. In addition, due to an oversight during the design of the NSTX upgrade, one of the inner poloidal field coils, specifically PF-1b, was thermally close-coupled to the horizontal section of the inboard divertor flange, which was intended to be heated with hot helium during bakeout. This led to a compromise between achieving adequate temperature for bakeout and protecting the electrical insulation of PF-1b. NSTX-U was unable to achieve a proper high temperature (350°C) bakeout of the inner divertor. The new design of the tiles and their mounting system increases the thermal heat flux and halo current capability of the divertor and the thermal decoupling of the inboard divertor from the PF-1b allows proper bakeout in preparation for high performance plasmas in NSTX-U during the next run period.

This work is supported by US DOE Contract No. DE-AC02-09CH11466

Eligible for student paper award?:

No

T.OA3: Blankets and Tritium Breeding: Liquid Breeders / 495

Integration of the Neutral Beam Injector System into the DCLL breeding blanket for the EU DEMO

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The integration of plant systems involving penetrations into the in-vessel components, like H&CD, fuel cycle and diagnostics, is a complex task constrained by top level requirements of remote maintainability and high reliability. Within the EUROfusion PPPT Program, some activities are ongoing to assess the integration of different systems into the breeding blanket, specifically NBI, ECRH launchers, diagnostics sightlines, fueling lines and specific protections for the FW (like start-up limiters).

This work describes the integration of the Neutral Beam Injector (NBI) system into the Dual Coolant Lithium-Lead (DCLL) breeding blanket for the EU DEMO. After identifying the major issues impacting the mechanical, thermal-hydraulic and neutronic behavior of the blanket, the integration efforts have been focused on minimizing the invasiveness of the NBI system and exploring different NBI options for the best compromise between plasma heating and breeding blanket performance. This paper describes the adaptation of the DCLL breeding blanket design to allocate the neutral beam duct. A particular attention is devoted to the redistribution of breeding and shielding functions, the new path of fluid circuits and the additional cooling needs.

The consequences of design modifications on key neutronic aspects like Tritium Breeding Ratio (TBR) and shielding capability are addressed. Besides, after a brief discussion regarding the thermal loads transferred to the breeding blanket walls from the neutral beam and the plasma, a preliminary thermal assessment of the proposed integration solution is presented.

Eligible for student paper award?:

No

T.OA2: Divertors and PFCs: Tungsten / 496

Investigation of ITER-grade tungsten under very high heat loads.

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Experiments carried out on advanced large tokamaks showed effective use of tungsten for making in-Vessel Components, interacting with the plasma. However, in reactor size fusion devices such as ITER and DEMO, are expected the critical loads on the divertor plates both in quasistationary stage and in pulsed events (disruption, VDE, ELMs et al.). High heat loads can cause not only increased erosion and destruction of material surface, but also strong absorption of tritium in erosion products. Usually, it's difficult to obtain divertor ITER-like power load in advance fusion devices with magnetic confinement. Therefore, to simulate ITER conditions powerful e-beam try to use, but it can't replace the real simulation by plasma. As example, JET and ASDEX-U experiments with movement of the molten W droplet, can be explained by electron emission from this droplet in the magnetic field.

On T-10 tokamak with a powerful ECR heating, were obtain regimes with nonambipolar energy flow on tungsten tiles of circular toroidal limiter. ITER-grade tungsten was use, which is intend for the ITER Dome divertor, manufactured by RFDA. The interiors of the limiter are heated to temperature exceed of 2000 0C and estimated heating power is more than 10 MW/m2. Spectroscopic line WI

near this plates show exponential increasing, but total radiation power decrease from 50% to 15%, and radiation loss at the boundary increase 3-4 time.

In this regime, there were deep and long cracks and powerful arcing occurred on W tiles. At that, cracks in ion side are perpendicular to tile edges and parallel to each other, as threads. The area of cracks coincide with the area of arcing. The edges of the cracks were melt and arc craters have been scattered not only across the surface but located along the cracks. All tiles surface was cover by resolidificated tungsten, on which there were many arc microcraters.

The report discuss the nonambipolar mechanism of energy flow on metal surfaces, leading to self-heating in the presence of arcs, the ecton mechanism of arcing, mechanism of cracking and estimation of tritium absorption in such kind of cracking.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 106 / 497

Designing for Tokamak Emergent Behaviour using a Hierarchical Systems Engineering Architecture Design Process

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Fusion reactors are complex machines in which many systems operate in concert to achieve the required behaviour. Controlled fusion is dependent on a fine balance between these systems, whose functions often overlap or conflict. A tokamak's functions can be described by simple terms like fuelling, heating, current generation or plasma stabilisation, but their technological realisation can be very complex, with different systems sharing responsibilities, interacting positively or adversely in a tightly controlled balance. Heating systems can be used to affect plasma current, divertors can contribute to plasma stability and fuelling systems can be used to help or hinder both. Tokamaks are designed on this basis because the performance required is at the very edge of known technological capabilities.

A tokamak can be viewed as a system dominated by "Weak Emergence", the definition of which is a system whose desired functions exist at a holistic abstract level, although its behaviour can in principle be reduced to its constituent parts. These systems are difficult to design because their properties cannot be easily understood in terms of the properties of their constituent parts alone, instead being dominated by dynamic interrelationships between parts during operation.

Faced with such complex interrelationships, designers can adopt a conservative approach and base their designs on minimal changes to an existing known design. This is not a viable option for the DEMO fusion power plant, which has to incorporate functions not present in research reactors, such as large-scale tritium breeding, whose enabling technologies will require periodic maintenance and renewal over their lifetime.

The Systems Engineering design process, as defined by ISO15288:2015, provides a systematic approach to the design of complex systems and their emergent behaviour. It divides the system design process into the development of a two stage, top-down hierarchy; an abstract, technologically agnostic "System Architecture" high level and a more practically oriented "System Design" lower level, with formal relationships between the two. The abstract representation allows the designer to develop and represent incremental insights obtained about the system's requirements and emergent behaviours, before progressing towards its technological realisation in the System Design. This process encourages a more pragmatic and less conservative approach to solution development and optioneering.

This paper presents examples of how the interactions of complex interrelated technological systems in the tokamak can be reinterpreted as an abstract System Architecture more closely related to the

desired emergent functions. Relationships between systems at a high level are represented by services and qualitative dependencies, rather than solely the flow of materials and energy. SysML is used as the representation language, encoding each system in terms of its statement of purpose, functions and performance parameters, operating within a formally defined context.

The paper shows how the complexities of the DEMO design can be managed without resorting to design conservatism. The System Architecture will be used to manage and structure the creative processes involved in the technological choices at the System Design level, providing a framework in which novel solutions and optioneering can take place, supporting the design innovations needed to bring DEMO to fruition.

Eligible for student paper award?:

No

W.OA1: Materials II / 498

Material Solutions for Flow Channel Inserts for Liquid Metal Blankets

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Flow Channel Inserts (FCI) are key components, originally proposed in European blanket conceptual designs, required in liquid metal-cooled and/or -bred fusion reactor blankets such as the Dual Coolant Lead-Lithium (DCLL) or Helium-Cooled Lead-Lithium (HCLL) concepts. The FCI minimizes the magnetohydrodynamic (MHD) pressure drop and enables higher outlet temperatures for liquid metal than temperature limits for the structural alloys. The primary functions of the FCI are to electrically insulate the flowing liquid metal from the conductive metal structures and to thermally insulate the structures from the high temperature liquid. The FCI must also provide structural integrity, allow adequate tritium breeding while operating in a harsh radiation, temperature gradient, and chemical environment.

Two FCI materials concepts are currently considered: SiC-based composites and metal-encapsulated insulators. SiC-based composites, in particular the continuous SiC fiber-reinforced SiC-matrix (SiC/SiC) composites, are the prime candidate due to the superior high temperature capability and anticipated radiation tolerance. These composites in the common two-dimensional fabric lay-up architectures have been shown to satisfy the general requirements for FCI in as-manufactured and fission neutron-irradiated conditions. However, a few critical design-limiting issues and uncertainties remain with regard to the use as an FCI in a fusion nuclear environment. The metal-encapsulated insulator concept is a potential near-term alternative to the SiC-based FCI concept. Although the concept presents more limitations on the operating temperature and neutron tolerance, proper choice of both the insulating and encapsulating materials may allow operations in less aggressive conditions. These FCI concepts are discussed in terms of the development status, design limiting issues, potential solutions, and the path-forward for technology development and qualification.

This work was supported by the Office of Fusion Energy Sciences, U.S. Department of Energy, under contract DE-AC05-00OR22725 with UT-Battelle, LLC.

Eligible for student paper award?:

No

T.OP2: Fueling, Exhaust, and Vacuum Systems / 499**Brief History and Status of Cryogenic Pellets in Fusion Energy Research**

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Brief History and Status of Cryogenic Pellets in Fusion Energy Research*

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High-speed injection of solid fuel was first proposed in 1954 (Spitzer et al., USAEC Report NYO-6047) as a possible solution to the problem of transporting fresh fuel across the confining magnetic fields into the plasma of a fusion reactor. While it took a few decades, the use of cryogenic pellets (typically H₂ and D₂) on fusion experiments eventually became commonplace, with most tokamaks and stellarators around the world equipped with a pellet injector at some point. Hydrogen pellet injection devices operate at very low temperatures (10 to 20 K) at which solid hydrogen ice can be formed and sustained. Most injectors use conventional pneumatic (light gas gun) acceleration to routinely accelerate macroscopic sized pellets (0.3 to 6 mm diameter) to speeds of 100 to 1000 m/s. Two other key operating parameters for plasma fueling are the pellet injection rate and time duration, with a single pellet adequate for some experiments and a steady-state injection rate of up to 50 Hz or greater appropriate for other experiments. Even testing with T₂ and D-T pellets were carried out in the late 80s to demonstrate reliable operations with the radioactive isotope. As a testimony to the maturity of this specialized technology, pellet injectors have been available commercially from a Russian company (PELIN) for about 20 years; several of these systems are presently operating on major fusion experiments in Europe and Asia. A significant finding from tokamak studies in the late 90s indicated that appreciable improvement in pellet penetration and fueling efficiency could be achieved by injecting pellets from the magnetic high-field side (HFS) of the device. This scheme requires the use of curved guide tubes to transport/deliver the pellets to the plasma. Extensive testing in the lab and pellet experiments on fusion devices around the world have shown that intact pellets can be reliably delivered through any curved tube track if the speed is maintained below a specific limit (typically 100 to 500 m/s) for the given system. Thus, pellet fueling from the HFS is planned for ITER. In addition to plasma fueling, cryogenic pellets have often been used for particle transport and impurity studies on fusion experiments, including Ne, Ar, CH₄, and gas mixtures. During the last decade or so, a few new applications for cryogenic pellets have been demonstrated, one for edge-localized mode mitigation, one for plasma disruption mitigation (requires large pellets that are shattered before injection into plasma), and another in which pure Ar pellets are used to trigger runaway electrons in the plasma for scientific studies. In the paper, a brief history and the present status of cryogenic pellets in fusion energy research will be presented.

- This material is based upon work supported by the U.S. Department of Energy, Office of Science, Office of Fusion Energy Sciences, under contract number DE-AC05-00OR22725.

Eligible for student paper award?:

No

W.OP2: Heating and Current Drive / 500

Smoothly Varying Injected Neutral Beam Voltage and Current Provides New Capability on the DIII-D Tokamak*

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A significant engineering upgrade of the DIII-D neutral beam system has recently been completed, providing the ability to smoothly and independently vary the beam voltage and current. This enables greater control of the injected beam power, torque, and plasma instability drive for fusion experiments. Modifications to the high voltage equipment and the Plasma Control System were made and tested over the last year to allow beam energy to vary by as much as 20 kV over a 0.5 sec period anywhere within the 45-85 kV operating range of the beams. The beam current can be made to track the voltage (keeping the perveance constant), or the current and voltage can be varied independently (scanning the perveance). Smooth variation of beam energy avoids the extremely perturbative effects of pulse width modulation, the only tool previously used for regulating the injected neutral beam power. With independent control of voltage and current, the beam ion velocity space can be tailored to facilitate new experiments that explore, for example, the detrimental effects of Alfvén eigenmodes. These modes are driven by energetic particles and can possibly be avoided in steady state scenarios by timely variation of the beam energy while maintaining constant input power. A description of the modifications made to the power supply and beam controls will be presented, as well as some initial physics results employing the new variable beam energy system.

* Support for design and implementation of this new operating mode for the neutral beam system was provided through General Atomics' Internal Research & Development. Testing on DIII-D was supported in part by the US Department of Energy, Office of Science, Office of Fusion Energy Sciences, using the DIII-D National Fusion Facility, a DOE Office of Science user facility, under Award No. DE-FC02-04ER54698.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 62 / 501

Neutronic study and shielding performance analyses for CFETR blankets

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The fundamental parameters calculations addressing tritium breeding ratio (TBR), neutron wall loading (NWL) and nuclear power generation on a Chinese Fusion Engineering Testing Reactor (CFETR) neutronic analysis model were performed using MCNP code to investigate the feasibility of the helium cooled pebble bed breeder blanket. The neutronic model was created as a 11.25° torus sector of tokamak manually based on the engineering data of one CAD program, including the first wall (FW), breeding unit, manifold, back plate, shield, vacuum vessel (VV), thermal shield and TFC. One major requirement of the machine is to provide sufficient protection for the vacuum vessel and superconducting components against the radiation penetrating in-vessel components and vessel. The neutron fluxes across the inboard torus mid-plane and the radiation hazard such as the accumulated displacement damage (DPA) and helium production in steel which may deteriorate the material performance were also calculated and presented in this paper. Results show that neutrons are attenuated and slowed down efficiently by components placed between plasma and TF coils. And the achievable results can mainly be acceptable.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 102 / 502

Final acceptance test 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 part of the ITER project.

The Ion Source and Extraction Power Supplies (ISEPS) for SPIDER are supplied by OCEM Energy Technology s.r.l. (OCEM) under a procurement contract with Fusion for Energy (F4E) covering also the units required for MITICA and ITER injectors.

ISEPS, with an overall power rating of 5 MVA, form a heterogeneous set of items, ranging from power transformers, medium voltage power distribution equipment at 6.6 kV to solid state power converters and including four 1 MHz radiofrequency generators of 200 kW output power. Both high voltage, down to -12 kV and high current, up to 5kA, power supplies are present.

SPIDER ISEPS has been installed in the NBTF SPIDER High Voltage (HV) Hall, on an air-insulated platform ("HV Deck"), at a nominal voltage to ground of -96kVdc.

The installation of SPIDER ISEPS started in June 2015 and was completed in September 2015. Functional checks started thereafter and power testing in January 2016. The formal Site Acceptance Tests (SAT), witnessed by F4E, Consorzio RFX and the ITER Organization were successfully completed in April 2016. However, few aspects requiring further testing were identified and brought to successful completion in September 2016. After the verification of the final documentation package and the conclusion of the acceptance process, the procurement was closed in February 2017.

This work will give an overview of the testing phase on Site summarizing the most interesting findings on the subsystems operation in the final installation conditions. In particular the focus will be on the final ISEPS acceptance tests, which allowed to prove the correct operation of the different ISEPS subsystems on dummy loads, under remote control of CODAS (Control and Data Acquisition System) and Interlock systems.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 24 / 503

Advanced Plasma Diagnostic Analysis using Neural Networks

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Machine learning techniques, specifically neural networks (NN), are used with sufficient internal complexity to develop an empirically weighted relationship between a set of filtered X-ray emission measurements and the electron temperature (Te) profile for a specific class of discharges on NSTX. The NN response matrix is used to calculate the Te profile directly from the filtered X-ray diode measurements which extends the electron temperature time response from the 60Hz Thomson Scattering profile measurements to fast timescales (>10kHz) and greatly expands the applicability of Te profile information to fast plasma phenomena, such as ELM dynamics. This process can be improved by providing additional information which helps the neural network refine the relationship between Te and the corresponding X-ray emission. NN supplement limited measurements of a particular quantity using related measurements with higher time or spatial resolution. For example, the radiated power (Prad) determined using resistive foil bolometers is related to similar measurements using AXUV diode arrays through a complex and slowly time-evolving quantum efficiency curve in the VUV spectral region. Results from a NN trained using Alcator C-Mod resistive foil bolometry and AXUV diodes are presented, working towards hybrid Prad measurements with the quantitative accuracy of resistive foil bolometers and with the enhanced temporal and spatial resolution of the unfiltered AXUV diode arrays.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 9 / 504

Steady State and Transient Thermal Analysis of the Updated Helium Cooled Solid Breeder Blanket for CFETR

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This paper presents the results of the steady state and transient thermal analysis of the updated helium cooled solid breeder blanket for Chinese Fusion Engineering Test Reactor (CFETR). The updated design of the helium cooled solid breeder blanket for CFETR has been described. The commercial finite element method code ANSYS is used for the thermal analysis in this work. Steady state thermal analysis of the updated blanket has been performed, showing that the temperatures of different materials of the blanket module are below the corresponding temperature limits. The three dimensional transient thermal analysis of the updated helium cooled solid breeder blanket for CFETR under normal pulsed operation has been conducted. The temperature of different components during flat-top burning time calculated by transient thermal analysis matches well with that of steady state thermal analysis. Furthermore, the transient thermal analysis of the blanket under LOFA, in-box LOCA and ex-vessel LOCA conditions are also performed and critically discussed.

Eligible for student paper award?:

No

T.PLN: Plenary T / 505

Progress in the EU DEMO Research and Design Activity

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As part of the Roadmap to Fusion Electricity, Horizon 2020, Europe initiated a pre-conceptual design study of a Demonstration Fusion Reactor Concept (DEMO) a few years ago, which targets the generation of a few hundred MW of net electricity and the demonstration of a closed tritium fuel cycle in the 2050s.

The design and R&D approach adopted include some distinctive elements such as: 1) a strong philosophy of integrated design at an early stage to encourage a more 'systems thinking' culture and to bring major clarity to a number of critical design issues and overall integration challenges; 2) an improved understanding of system context as a foundation for informed plant design concept and technology development programmes; 3) a prudently modest extrapolation from the ITER physics and technology basis, in order to minimize programme/development risks and their associated mitigation costs; 4) multiple DEMO plant design architectures are studied in parallel (e.g. reactor configurations such as a double-null tokamak), as are major sub-systems or technologies for which there are particularly high technical risks or low maturity (e.g. the divertor, the breeding blanket, etc).

The progress of the EU DEMO design and R&D activities to date is described, with a focus on the areas that are believed to have a strong hand in defining the conceptual layout of the DEMO device, and drive its performance. Recently, a number of external and internal developments have occurred that challenge some of the assumptions underpinning the original schedule. This includes the delay of ITER construction and DT operations and a greater appreciation of the 'integration challenge' required to define a robust plant architecture. A reasonable extrapolation from ITER results is maintained, and a provisional, updated DEMO schedule is discussed.

The pulsed EU DEMO baseline design point continues to be the primary configuration studied (in particular for integration issues –many of which have broad applicability to other reactor designs); however a number of alternative reactor configurations are now also being studied in earnest. These include for example a double-null divertor machine, and a pulsed "flexi-DEMO" machine capable of transitioning to steady-state operation. Preliminary results of studies exploring the available design space and defining the main parameters and technical characteristics for these configurations are shown. The design strategy of the plasma-facing system is discussed, and the preliminary definition of a DEMO plant layout is presented, aimed at enabling further design integration studies as well as safety and cost analyses for the wider plant auxiliary systems.

Design and technology down-selection will be of vital importance on the path reaching a DEMO concept and it is critical that a robust decision-making framework is established in the years to come to support future decisions. Thoughts on such a framework are presented here, and on its application to the fusion R&D programme in the future to progressively narrow down sub-system technologies and reactor architecture options.

Eligible for student paper award?:

No

R.OP4: Stellarators / 506

Review of Research and Engineering on the H-1 Helic

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The H-1 flexible heliac is a medium-sized helical axis stellarator of major radius $R=1\text{m}$, and average minor radius $\langle r \rangle \simeq 0.15 - 0.2\text{ m}$. Precise control of the ratio k_h of the helical winding current to the ring coil current provides access to a wide range of magnetic configurations. This provides rotational transform $\bar{\iota}$ in the range $0.9 < \bar{\iota} < 1.5$ for $B_0 < 1\text{ T}$, with both stellarator-like and tokamak-like shear, and magnetic well from $\simeq 5\%$ to -2% (i.e. hill).

The coil-in-tank construction technique allowed the TF, ring, helical and inner VF conductors to be assembled on a 4m ϕ precision rotating table for accurate positioning using the measured magnetic axes of the electromagnets. A recent tomographic electron-beam magnetic field mapping exercise has shown that the location accuracy after installation and years of use is $< 2\text{mm}$. Under excitation there is a small distortion of the support structure, which is designed to resist overturning and other forces $\sim 5\text{ Tonne}$ per TFC. Worst case displacements of $\sim 2\text{mm}$ and rotational transform change of -0.5% are found, extrapolating to -2% at 1T .

Fluctuations ranging from drift waves to Alfvénic have been observed in the $1\text{-}100\text{s}$ of kHz range. Propagation, dispersion and fluctuation-induced transport data will be presented, including synchronously imaged 2D reconstructions of density fluctuations. Comparison of the profiles with 3D compressible MHD predictions (CAS3D) show best agreement with a beta induced Alfvén eigenmode. Density fluctuation profiles are also observed by a 21 channel 2mm interferometer using imaging techniques to improve resolution. Recently a third current controller has enabled automated two-parameter configuration scans of rotational transform and magnetic well, revealing a general increase in fluctuations as the well depth is decreased.

Several unique optical diagnostics have been developed on H-1 and deployed on larger international devices. Coherence imaging is a very high resolution interferometric technique used initially on H-1 to determine ion temperature and flow by the Doppler effect. Further developments include isotope ratio imaging and polarisation effects. Imaging in synchronism with the applied RF heating has revealed images of RF propagation in H-1, suggesting an electrostatic mode clearly dependent on ω/ω_{ci} . Tests of a recently installed ‘compact Alfvén’ antenna and adjustable side shields are underway. Results from recent investigations will be presented together with selected results from the 25 year history of H-1.

Eligible for student paper award?:

No

R.OP4: Stellarators / 507

The quasi-optical steady state 10 MW ECRH system of Wendelstein 7-X - commissioning, plasma operation and future plans

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During the first operational phase (OP1.1) of Wendelstein 7-X (W7-X) electron cyclotron resonance heating (ECRH) was the exclusive heating method, and it will also remain dominant in the next operational campaign (OP1.2). The microwaves are provided by ten gyrotrons (140GHz, 1MW), which

were developed within the PMW-project at KIT. The ECRH-system is also the only heating system able to operate steady state at W7-X. The quasi-optical transmission line to the plasma vessel is the first of its kind. It consists of ten single beam sections (SBS) matching the non-perfect Gaussian beam output of the gyrotrons to the subsequent 40m long multi-beam section (MBS), which has an efficiency of at least 97%, and is intrinsically broadband. The whole MBS consists of two lines, each with 7 large-area multi-beam mirrors transmitting five of the ten gyrotron beams in parallel. Finally, the MBS separates the beams and directs them to their respective vacuum window in a single beam line. The windows interface to the four equatorial launchers installed on W7-X with three beam lines each. The plasma facing mirror of each beam line is steerable in the poloidal and toroidal direction. This way, the radial heat deposition in the plasma and the plasma current drive can be varied, respectively.

The flexibility to adjust the mirrors of the quasi-optical transmission line enables alignment of the beam line with the high-power microwave beam itself. Furthermore, only two beams were necessary for the final adjustment of the imaging MBS, demonstrating its nearly perfect imaging properties. Finally, the overall transmission efficiency up to the launcher window was determined to be 94% including diffraction, beam truncation, misalignment and absorption of the mirrors and the atmosphere.

The whole ECRH-system was successfully used in OP1.1 for plasma start-up, wall conditioning, heating and current drive - all disciplines with high reliability. However, only six gyrotrons with a total power of maximum 4.3MW were used, because the divertor was not yet installed in this commissioning phase of W7-X. Therefore, the energy throughput during a discharge was limited to 4MJ. Nevertheless, electron temperatures up to $T_e \approx 10\text{keV}$ were easily achieved using ECRH with typically peaked profiles in case of central X2 deposition. The electron density was operated up to $n_e \approx 0.3 \cdot 10^{20} \text{ m}^{-3}$, leading to an ion temperature of 2keV. The ions are heated by electron ion collisions. Therefore, the T_i profile is determined by the - typically flat - electron density profile of the electrons. Even though the densities in OP1.1 were far away from the X-mode cutoff at $n_{x-cutoff} = 1.2 \cdot 10^{20} \text{ m}^{-3}$, a multi-path O2-heating scenario could already be demonstrated. It is thus feasible to achieve almost full equilibration of electrons and ions during $n_e > n_{x-cutoff}$ scenarios in OP1.2. All ten gyrotrons will be in operation by then, with a total power of up to 9MW and a typical pulse length of 10s -100s. Furthermore, two so-called remote-steering launchers (RSL) were installed to investigate advanced current drive scenarios, and to test this reactor relevant launcher concept at high-power.

Eligible for student paper award?:

No

R.OP3: Tritium Extraction and Control / 508

Estimation of tritium release and permeation behavior in water cooled solid breeder blanket

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Tritium release behavior from solid breeder pebbles in a blanket module is strongly influenced by not only diffusion in grains but also surface reactions. A part of tritium released to a purge gas permeates into a coolant. Nishikawa et al. proposed the tritium release model from solid breeder materials that includes tritium diffusion and surface reactions such as isotope exchange reaction, water form reaction and absorption/desorption of water vapor on grain surface [1]. Recently, tritium release behavior from Li₂TiO₃ pebbles by DT neutron irradiation was experimentally investigated by Edao et al. and a ratio of HT and HTO in released tritium was reported [2]. On the other hand, the release behavior of water vapor from Li₂TiO₃ pebbles manufactured by the same method was reported [3]. However, the relationship between HT/HTO ratio and the release behavior of water

vapor has not been discussed yet.

In this work, numerical calculation of tritium release behavior from Li_2TiO_3 pebbles was performed applying Nishikawa's model and calculation results were compared with experimental data such as [2]. Then, it was shown that HT/HTO ratio is relatively influenced by isotope exchange reaction rate and the amount of released water vapor generated by Li_2TiO_3 pebbles and H_2 in the purge gas. Additionally, tritium behavior in a water cooled solid breeder blanket module was numerically estimated considering tritium permeation loss into cooling water. Tritium mass transfer parameters such as diffusivity and isotope exchange reaction rate were referred from literature data for several kinds of solid breeder material such as Li_2TiO_3 and Li_4TiO_4 . The influences of temperature in the solid breeder pebble bed and grain size of solid breeder on chemical form of released tritium were quantitatively shown. Tritium inventory in grain bulk, grain surface, purge gas, cooling pipe or cooling water was estimated respectively. Tritium inventory in grain bulk was largest among these components.

[1] M. Nishikawa et al., *J.Nucl.Mater.*, 325 (2004) 87-93.

[2] Y. Edao et al., *Fusion Eng. Des.*, 112 (2016) 480-485.

[3] R. Yamamoto et al., "Li mass loss from Li_2TiO_3 with excess Li pebbles fabricated by optimized sintering condition", SOFT2016, Sep.5-9, 2016, Prague, Czech Republic.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 112 / 510

Hydrogen isotope permeation through tungsten deposition layer formed on Ni plate by plasma sputtering method

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Understanding of tritium behavior in the plasma facing wall of a fusion reactor is important from viewpoints of fuel control and tritium safety. Tungsten is a primary candidate of plasma facing material because of low sputtering rate and low tritium solubility. However, even if tungsten is used on plasma facing wall, formation of deposited layer cannot be avoided in long-term operation of the fusion reactor. Because the structure of deposited layer is greatly different from the structure of original material, tritium behavior has to be understood for not only tungsten bulk but also tungsten deposition layer. In this work, hydrogen or deuterium gas driven permeation through tungsten deposition layer was investigated. Then, diffusivity of hydrogen isotope in tungsten deposition layer was estimated by numerical fitting method with considering an influence of Ni substrate although the influence of Ni substrate was ignored in our previous work 1.

Samples of tungsten deposition layer were formed on circular substrates of nickel by hydrogen plasma sputtering. The thickness of the deposition layer was evaluated to be about 700 nm against 20000 nm in thickness of Ni. The deposition layer with Ni substrate was clamped between a copper gasket and a stainless steel flange. In experiment A, the secondary side was closed in vacuum and hydrogen gas was supplied in the primary side and then the hydrogen permeation flux was obtained from the pressure rise at the secondary side. In experiment B, the secondary side was continuously evacuated and hydrogen or deuterium was supplied in the primary side. The permeated hydrogen isotopes were measured by a quadrupole mass spectrometry. The temperature in these experiments were set from 200oC to 500oC.

The pressure rise with time in experiment A was analyzed by TMAP calculation and diffusivity was evaluated. The obtained hydrogen diffusivity was slightly larger than that obtained by ignoring Ni substrate. It was found that hydrogen diffusion in Ni substrate affects slightly hydrogen permeation

rate. The absolute value of hydrogen diffusivity was much smaller than that in tungsten bulk. In experiment B, when deuterium gas was supplied in the primary side after hydrogen permeation experiment was finished, not only deuterium but also hydrogen were appeared in the secondary side. This result indicates that hydrogen trapped in the sample was released by the isotope exchange reaction with deuterium migrating in tungsten deposition layer. Deuterium gas soak is considered to be effective for tritium recovery from tungsten deposition layer.

1 K. Uehara et al., Fusion Eng. Des. 98-99 (2015) 1341-1344.

Eligible for student paper award?:

Yes

T.OA2: Divertors and PFCs: Tungsten / 512

Precipitation of transmutant elements in neutron irradiated tungsten

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As the leading plasma facing material in fusion reactors, tungsten is confronted with extremely hostile environment, characterized by high temperature, and high fluxes of heat and particles (i.e., D, T, He, and neutrons). One of the primary concerns is the generation of transmutation elements (i.e., Re, Os) and the subsequent radiation-induced segregation and precipitation, and the resulting thermo-mechanical property degradation induced by the 14 MeV-peak neutron irradiation. In this study, we have used advanced electron microscope methods to explore the response of tungsten to high dose neutron irradiation in the High Flux Isotope Reactor, focusing on the detailed characterization of irradiation-induced W-Re-Os precipitates in polycrystalline tungsten through TEM, X-ray mapping in STEM, multivariate statistical analysis data-mining of the X-ray data and transmission Kikuchi diffraction. The association of voids and precipitates, the chemical compositions, crystal structures and phases of precipitates along the grain boundary and within the grains were identified. The results showed that the intragranular precipitates are sigma-phase while the precipitates along the grain boundaries are chi-phase. The kinetics process of transmutant elements and radiation defects were briefly discussed to reveal the formation process of the observed precipitates.

In addition, we also investigated the hardening contribution of W-Re-Os precipitates. A dispersed barrier hardening model informed by the available microstructure data was used to predict the hardness. The results indicated that the formation of intermetallic second phase precipitate dominant the radiation-induced strengthening with a relatively modest dose (>0.6 dpa). The hardening strength factor of the transmutation-induced precipitates was also determined to be 0.6.

The work presented in this paper was partially supported by Laboratory Directed R&D funds at ORNL. The research was also sponsored by the US Department of Energy Office of Fusion Energy Science under grants DE-AC05-00OR22725 with UT-Battelle LLC and by the US-Japan PHENIX project under contract NFE-13-04478, with UT-Battelle LLC.

Eligible for student paper award?:

No

M.OA2: Divertors and High Heat Flux Components / 513**Modeling and Experimental Validation of Physics Enabled by W7-X Scraper Element Divertor Components****Author:** Jeremy Lore¹**Co-authors:** A. Lumsdaine¹; F. Effenberg; G Douglas Loesser²; G. Wurden; H. Frerichs; Han Zhang; Hauke Hoelbe³; J.H. Harris; Jean Boscary; Joachim Geiger⁴; Joris Fellingner⁴; L. Stephey; Marc Sibilia⁵; Marcin Jakubowski⁶; Peter Titus⁷; the W7-X Team¹ Oak Ridge National Laboratory² PPPL³ Hamburg University (DE)⁴ Max-Planck-Institut für Plasmaphysik⁵ Princeton Plasma Physics Lab.⁶ IPP⁷ Princeton Plasma Physics Lab**Corresponding Authors:** dloesser@pppl.gov, lorejd@ornl.gov, hauke.hoelbe@cern.ch, marcin.jakubowski@ipp.mpg.de, msibilia@pppl.gov, hzhang@pppl.gov, jean.boscary@ipp.mpg.de

A set of add-on components referred to as Scraper Elements (SE) were designed as a passive solution to a predicted heat flux overload of certain areas of the main Wendelstein 7-X (W7-X) stellarator divertor during long-pulse operation. W7-X will soon begin its first phase of operation using the first set of plasma facing components (PFCs) and a magnetic topology to realize an island divertor configuration (OP1.2). In an island divertor the core plasma is surrounded by an island chain with a helicity determined by the edge value of the rotational transform. The island chain is intersected by the PFCs, leading to heat and particle fluxes that typically manifest as a set of stripes with neutral baffling to guide recycled particles into pumping volumes. One challenge associated with stellarator island divertor configurations is to keep the edge rotational transform constant to maintain the desired topology. A net toroidal current evolving during a discharge will modify this transform unless it is opposed using applied driven current (e.g., ECCD) or by changing the toroidal field to maintain a fixed value. This issue is mitigated in W7-X as one of the optimization goals targeted during the design process was a low bootstrap current. However, this property is persistent only in certain configurations. In some long-pulse configurations of interest the steady-state toroidal current is predicted to be sufficient to modify the edge transform by ~10%. Such a current and thus boundary topology evolution, without mitigation, would sweep heat flux across regions of the divertor with a reduced rating, resulting in transient overload before the island divertor configuration is restored in steady-state.

We designed the SE as a passive solution to this problem. In the long-pulse, high-power operational phase (OP2) ten SE, one for each divertor unit, would be installed to intercept heat flux to the overloaded areas during the transient phase while receiving a load less than its 20 MW/m² rating. Due to geometric limitations on the design, the SE continue to receive loading during the steady-state configuration, possibly leading to deleterious effects on pumping of neutral particles and impurity influx into the core plasma. To test both the positive and negative impacts, two inertially cooled SE will be installed in the middle of the OP1.2 campaign. Experiments both before and after the installation of SE will be used to determine if the predicted overload will exist, whether the SE protects the affected areas, and to assess any side effects associated with the SE. Special configurations will be used to mimic the effect of the OP2 configuration evolution, which is not directly accessible in OP1.2. The modeling associated with the design will be presented, along with the plans for validation using the W7-X diagnostic set. Details of the inertially cooled SE design and the finite element modeling performed to determine power and pulse length limitations will also be shown. Support from D.O.E. contracts DE-AC05-00OR22725, DE-AC52-06NA25396, DE-AC02-09CH11466.

Eligible for student paper award?:

No

W.OP1: Magnets / 514**Fabrication Status of ITER Central Solenoid Modules**

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The fabrication of the modules for the ITER Central Solenoid (CS) is in progress at General Atomics (GA) in Poway, California, USA. This purpose built facility has been established with the requisite tools and machines to fabricate the seven 110 tonne CS modules (six required plus one spare). GA's project responsibility is completing the fabrication design, developing and qualifying the fabrication processes and tools, and fabricating the seven modules, and testing the modules at 4.7 K with full current. The current schedule has the first module's fabrication completing in 2018 followed by electrical insulation and full current testing. Upon completion of testing, the modules will be shipped to the ITER site for assembly into the CS stack.

The Central Solenoid is a key component of ITER, providing the inductive voltage to initiate and sustain the plasma current, and position and shape control of the plasma. When completed, the CS is a 900 tonne assembly, which will be lowered as a single unit into the core of the tokamak. The design of the CS has been a collaborative effort between the US ITER Project Office, the international ITER Organization and GA

GA has completed the fabrication of a qualification coil, simulating all of the processes and exercising all of the tooling required to fabricate a production module. This qualification coil was recently completed and will undergo a cool down cycle for verification of the electrical insulation and to commission the final test station. Many lessons were gained from the production of the qualification coil and these were incorporated into the module fabrication processes.

GA currently has three production modules in fabrication, all in different stages of the process. The first module is currently being insulated, a process that requires more than 4 months. The second module is completely wound and the seven segments are being joined together into a continuous 6km long conductor. For the third module, the 900 and 600m lengths of conductor are currently being wound into six and four layer pancakes respectively.

The fabrication process for a module is approximately 22 months, start to finish, and followed by five months of testing, which includes preliminary electrical testing followed by high current (48.5 kA) tests at 4.7K and concluding with final electrical tests including a room temperature Paschen test at 30kV. The first module undergoes critical current sharing temperature measurements as well. This presentation describes the processes and status of the fabrication and testing of the CS Modules for ITER.

This work was supported by the US DOE Energy Office of Science and UT-Battelle/Oak Ridge National Laboratory under DE-AC05-00OR22725 and 4000103039.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 113 / 515

TCAP hydrogen isotope separation process under development at ICSI Rm. Valcea

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A pilot plant for tritium removal from tritiated water is in operation at ICSI Ramnicu Valcea and is based on catalytic isotopic exchange between tritiated water and hydrogen/deuterium followed by cryogenic distillation aiming to recover tritium. A cryogenic distillation cascade consisting of four distillation columns is in operation and significant effort is required in various batch mode operations for achieving high tritium concentration. The main drawback of the cryogenic distillation cascade that is the tritium hold-up, may be overcome by complementing the cryogenic cascade with a thermal cycling adsorption process. The main references of the mathematical model will be presented together with some references for the key components, such as the adsorbent. A comparative evaluation of various adsorbents will be given having in view the trade-off between the adsorption properties and the thermal properties of various adsorbents that have potential for implementation in such a process.

The process will be developed and provisions will be considered in view of scaling up for large throughputs having as ultimate goal the implementation in the Tritium Removal Facility from NPP Cernavoda. The developments are also relevant for the fusion activities such as ITER and DEMO.

Eligible for student paper award?:

No

M.OA3: Inertial Fusion Engineering and Alternate Concepts / 516

Status of the ICF program in China

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The inertial confinement fusion (ICF) program in China is to perform thermonuclear ignition and burning, and has made great progress so far. For ICF drivers, the laser facilities of SG-IIU (upgrading) and SG-IIIP (prototype) with both 8 beams and total laser energy output of tens kJ for 0.35 μ m wavelength (same below) and of SG-III with 48 beams and total laser energy of \sim 200kJ are operating and serving for target physics experiments. The ignition facility with laser energy output of more than 2.0MJ has been considered. The SG-III are very important to provide a scale-up from physical experiments in laser energy of hundreds kJ to MJ ignition. In addition, we are deeply investigating the target physics toward ignition. The innovative ignition schemes different from that performed on NIF, where the fusion ignition was unsuccessful, have been proposed as well.

Eligible for student paper award?:

No

W.OA2: Divertors and PFCs: Liquid Metals / 517

Testing Liquid Metal/Capillary Porous System Concepts as alternative solution for the Divertor target design of a Fusion Reactor in TJ-II.

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The use of liquid metals as plasma facing components (PFCs) in a future fusion reactor has been proposed as an alternative to solid metals, such as tungsten and molybdenum among others [1]. The expected advantages for the power exhaust issues, mainly arising at the divertor target at power densities of 10–20 MWm⁻², rely on the self-healing properties of liquid surfaces as well as the ability to in situ replacement of the surfaces exposed to the plasma by the effect of capillary forces (CPS design, [2]). Among the possible liquid metals (LM) presently considered as candidates for the development of an alternative solution to the Power Exhaust Handling in a future Fusion Reactor (Li, Sn, Ga), tin lithium alloys offer unique properties in terms of evaporation, fuel retention and plasma compatibility. This is the reason why this particular LM was chosen as main candidate in the US APEX project [3]. Very recently, LiSn (20-30:80-70at.%) alloys have been exposed to ISTTOK and TJ-II and very promising results on D retention and surface segregation of Li were obtained [4,5]. Motivated by these results a full campaign of comparative Li/ LiSn/Sn testing in TJ-II plasmas has been initiated. Solid and liquid samples have been exposed and a negligible perturbation of the plasma has been recorded in the Li and LiSn cases, even when stellarator plasmas are particularly sensitive to high Z elements due to the tendency to central impurity accumulation. The surface temperature of the liquid metal/CPS samples (made of a Tungsten mesh impregnated in SnLi, Sn or Li) has been measured during the plasma pulse with ms resolution by pyrometry and the thermal balance during heating and cooling has been used to obtain the thermal parameters of the LM/CPS arrangements as well as to calculate the thickness of the film interacting with the plasma. Temperatures as high as 1100K during TJ-II plasma exposure were observed for the LiSn case and hints of sputtering-enhanced evaporation were deduced from the temperature dependence of the lithium fluxes entering the plasma.

In this presentation a full account of the results obtained and their implications for the use of LM/CPS concepts in a future Fusion Reactor will be addressed. Particular attention will be given to the optimization of the thermal properties of the proposed designs.

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[2] Mirnov S V et al 2011 Nucl. Fusion 51 073044

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[4] J. Loureiro et al. Fus. Eng. And Des. (2017)

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<http://dx.doi.org/10.1016/j.nme.2016.11.026>

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Eligible for student paper award?:

No

W.OP1: Magnets / 519

Evaluation of ITER CS Module OD Band Structure for Lead Supports

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The central solenoid modules for the ITER project are powered by a complex system of busbars, feeders, lead extensions, electrical joints and integral coil terminals. The leads carry time-varying mod-

ule currents through predominantly poloidal fields, and thus develop substantial toroidally-oriented, cyclic, Lorentz loads. The support of such loading requires a robust structure which can function reliably over the design life of the tokamak. A unique design scheme, utilizing a thin band of Nitronic 50 stainless steel around the outer diameter (OD) of the coil, reacts and resists the Lorentz forces from the leads. To this end, various structures (i.e., double-lead cassettes and long-lead support channels) are attached to this 1.6 mm thick OD band. Numerous electrical breaks minimize eddy currents from transient fields but do not diminish the structural capacity of the band. All structural aspects of the OD band assembly are evaluated and qualified relative to ITER magnet structural design criteria (MSDC) requirements; static, fatigue and fracture. Flaws size requirements are specified at various locations with the smallest allowable flaws located at the edges of helium outlet penetrations and welded joints connecting OD band panels. In addition, the effects of heating from eddy currents on OD band temperature and thermal ratcheting from successive plasma pulses are addressed with multi-physics analysis.

Eligible for student paper award?:

No

R.OP2: PMI and Plasma Edge Physics / 520

Overview of plasma surface interactions in tungsten with helium plasma exposure

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The performance of plasma facing components (PFC) is one of the main issues facing ITER and future magnetic fusion reactors. Tungsten will be used in ITER as the PFC material and is considered to be one of the primary candidates for future reactors. However, recent experiments that exposed tungsten to He plasma exposure or He ion irradiation with ion energy less than about 100 eV (well below the threshold energy for physical sputtering or Frenkel pair production in tungsten) reveal significant surface modification, including the growth of nanometer-sized “fuzz”, and formation of a layer of nano-bubbles in the near-surface region [1,2]. It is widely accepted that He atoms in tungsten, like in other metals, are insoluble and tend to form small clusters, which serve as the nucleating event for the formation of larger gas bubbles. It is also clear from atomistic simulations [3,4] that the processes of trap mutation produce W interstitial atoms that lead to surface morphology modification as the interstitials diffuse to and annihilate at the surface, in addition to plastic flow and dislocation loop punching processes driven by high compressive stresses caused by over-pressurized clusters, or nanometer-sized bubbles, and these processes can alter both the tungsten surface morphology and the He clustering dynamics.

One of the challenges with describing these effects for the large-extrapolations in performance required for the PFCs in next-step devices beyond ITER is the large span of spatial and temporal scales of the governing phenomena and, therefore, the theoretical and computational tools that can be used. Fortunately, recent innovations in computational modeling techniques, increasingly powerful high performance and massively parallel computing platforms, and improved analytical experimental characterization tools provide the means to develop self-consistent, experimentally validated models of plasma materials interactions that govern the performance and degradation of the divertor and PFCs in the fusion energy environment. This presentation will describe the challenges associated with modeling the performance of divertor PFCs in a next-step fusion materials environment, the opportunities to utilize high performance computing and present examples of recent progress to investigate the dramatic surface evolution of tungsten exposed to low-energy He and H plasmas, as well as the coupled He-defect evolutions in bulk structural materials exposed to high-energy He and neutron irradiation before laying out a vision for developing a computational materials modeling framework for fusion materials behavior.

- 1 S. Takamura, et al., Plasma Fusion Res. 051 (2006) 1
 [2] M. J. Baldwin and R. P. Doerner, Nucl. Fusion 035001 (2008) 48
 [3] F. Sefta, et al., Nucl. Fusion 073015 (2013) 53
 [4] A. Lasa, et al., NIMB 156-161 (2013) 303

Eligible for student paper award?:

No

T.OA3: Blankets and Tritium Breeding: Liquid Breeders / 521

WCLL breeding blanket design and integration: lessons learned in 2016 and follow-up

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The Water-cooled lithium-lead breeding blanket (WCLL) is a promising viable option for European DEMO nuclear fusion reactor. The liquid lithium-lead is the breeder-multiplier flowing at low velocity and low temperature (i.e. about 330°C). Pressurized water is in charge to cool the structure and to transport the heat towards the power conversion system (PCS) and the energy storage system (ESS). The structural material is the EUROFER. The WCLL breeding blanket studied during 2016, in the framework of EUROfusion Project, is based on the single module segment approach. Basically, it is a breeder unit, which is repeated along the poloidal direction. The power is removed by means of radial-toroidal (i.e. horizontal) water cooling tubes in the breeding zone. The lithium-lead flows in radial-poloidal direction. A 100 mm thick plate will connect the breeding blanket segment with the vacuum vessel (VV), through an attachment system. All these components shall be designed to withstand the loads during normal operation and accidental conditions. Water and lithium lead manifolds are designed and integrated with a consistent primary heat transport system and the lithium lead system.

The paper discusses the WCLL breeding blanket design features through selected relevant thermo-mechanic, thermo-hydraulic and neutronic analyses. The lessons learned from the design review will be pointed out in order to present the reasons for the improvements and the needed analyses.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 66 / 522

Multi-design Innovative Cooling Research & Optimization (MICRO): a novel set of optimized solutions for enhanced heat transfer in DEMO

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Several novel design solutions for high performance cooling systems have been developed and realized by Consorzio RFX, permitting to experimentally simulate the challenging heat transfer conditions foreseen in the future fusion devices. The project, called Multi-design Innovative Cooling Research & Optimization (MICRO), has the triple objective to verify the present solution applied inside the MITICA experiment, to perform improvements with an acceptable pressure drop and reliable manufacturing process and to develop further optimized solutions with a detailed exploration of the Design Space investigating the interrelated effects of input geometric parameters.

The main advantages rely on the possibility to extend the fatigue life-cycle of different high thermal stress components and to investigate the possibility to employ alternative dielectric fluids instead of water.

Design solutions characterized by a large enhancement of the heat transfer process would in fact allow the exploitation of less performing fluids in terms of cooling capability.

If the unavoidable deterioration of the cooling parameters would not prevent satisfying the thermo-structural requirements set for such kind of components, these dielectric fluids would represent a significantly advantageous option with respect to the existing technologies. This is particularly relevant in view of DEMO and future power plants characterized by higher efficiency and reliability.

The numerical investigation has been carried out on a set of different scale components, focusing the CFD optimization stage on a suitable portion of the domain (exploiting the spatial repetition of the applied heat loads), while the numerical assessment from the mechanical point of view has been performed on a full-scale model in order to take into account the global effect of thermal strain and stress.

This paper gives a detailed description of the analyses performed with ModeFRONTIER, the optimization multi-objective software, together with the samples manufacturing and of the experimental tests that have been carried out so far.

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

Eligible for student paper award?:

Yes

W.POS: Poster Session W - Board: 110 / 524

An Equation of State and Compendium of Thermophysical Properties of Liquid Tin, a Prospective Plasma-Facing Material

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One of the engineering challenges of magnetic confinement fusion is handling the high particle and heat fluxes incident on the divertor. Solid plasma facing materials must withstand these difficult conditions with a minimum of erosion, melting, and cracking. Liquid metals as plasma facing materials potentially alleviate all of these concerns, and are therefore being actively investigated for that purpose.

With an eye toward assessing the impacts of different liquid metal PFCs beyond the plasma-material interface (i.e. on the tokamak and ancillary systems as a whole), some additions must be made to codes such as RELAP5 and MELCOR for Fusion that are used to perform thermal hydraulic and safety analyses of such systems. These include an equation of state and other thermophysical properties of the candidate liquid metals.

Lithium is a primary candidate liquid metal PFC, and past interest in lithium as a tritium breeding material has resulted in comprehensive physical property summaries, including development of an equation of state subsequently implemented in both RELAP5 and MELCOR for Fusion. Tin is an interesting alternative; though it has a higher atomic number ($Z = 50$), it has a suitably low melting temperature (232 deg:C), low vapor pressure, and lacks the high chemical reactivity and high tritium solubility of lithium.

We collect and summarize here the available measured thermophysical property data on liquid tin, including density, specific heat, sound speed, vapor pressure, thermal and electrical conductivity, viscosity, surface tension, and tritium solubility. We use the thermodynamic property data (density, specific heat, sound speed, and vapor pressure) to develop an equation of state for liquid tin that accurately predicts these measured properties, and discuss its implementation in MELCOR for Fusion.

Eligible for student paper award?:

No

W.OA2: Divertors and PFCs: Liquid Metals / 528

A Review of Recent Studies on Liquid Metal Plasma-Facing Components

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Power and particle handling in the plasma edge region is one of the key issues, affecting the successful operation of a steady state magnetic fusion power reactor. Tungsten has widely been used for plasma-facing components in existing fusion experiments and is envisaged to be employed for the ITER divertor, perhaps with seeded impurity for radiation detachment. Unfortunately, conventionally available tungsten is known to suffer from cracking due to its exceptionally high DBTT (for the Ductile-Brittle Transition Temperature). Most recently, nonetheless, efforts have been devoted to develop ductile tungsten and also W-W composite materials.

To resolve the mechanical property issue with solid divertor materials, over the past decade the use of liquid metals has been proposed and implemented in a number of medium-sized confinement devices. Experimental data so far have been encouraging with improved confinement performance. However, there are tremendous uncertainties and yet-to-be explored nature about the behavior of free-surface liquids and vapors, interacting with the edge plasma, particularly under off-normal

conditions such as disruption. Fluid dynamics simulation has begun only recently to understand the effect of liquid convection on hydrogen recycling, for example.

Presented in this paper is a review of the recent work on the interactions between plasmas and liquid metals such as molten lithium, the most widely used for plasma-facing components in magnetic fusion experiments, and a future perspective of the application of liquid metals for future fusion power reactors.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 38 / 529

CURRENT STATUS OF THE EU DEMO VACUUM SYSTEMS

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The European DEMO programme is running an activity which aims to develop a self-consistent and fully integrated design of the Tritium, Matter Injection and Vacuum Systems, supporting a tokamak operation with a high burn-up fraction. The architecture of the DEMO fuel cycle is mainly driven by the need to reduce the tritium inventory in the systems to an absolute minimum. This requires the continual recirculation of gases in loops without storage, avoiding hold-ups of tritium in each process stage by giving preference to continuous over batch technologies.

To meet these requirements for the vacuum systems, the Direct Internal Recycling Concept (DIR) has been elaborated which separates a DT fraction from the exhaust gas and continuously feeds the matter injection systems with DT that is not being cycled through the tritium plant. Furthermore, in the technical realization of the DIR concept - the so-called KALPUREX[®]-process - the cryogenic pumps typically used in existing devices for torus exhaust pumping have been replaced by (i) metal foil pumps that provide the exhaust gas separation needed, and by (ii) continuously operating vacuum pumps (vapour diffusion, liquid ring) based on mercury as perfectly tritium-compatible working fluid.

This paper describes the current status of the various elements in the vacuum system development programme. The research is focused on modelling and on associated experiments for code benchmark and design support. Recent experimental results gained from mercury liquid ring pump testing and from metal foil characterisation are presented and discussed. Two alternative and fall-back technologies such as a novel multi-stage cryopump concept with gas separation and NEG pumping based on new high capacity getters will be explained.

Eligible for student paper award?:

No

M.OP2: Materials I / 530

Application of Materials Science Advances to Fusion Energy

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The development of practical fusion energy as a commercial energy source is widely acknowledged as one of the greatest scientific and technical challenges for the 21st century. Due to the extreme operating conditions in the first wall and blanket regimes, utilization of very high performance materials is vital for achieving a viable cost-competitive fusion reactor design. The materials to be utilized in ITER are based on relatively conservative engineering designs and 1990s-era (or earlier) materials. Many of the current reference materials for the structure and functional applications of proposed DEMO fusion reactors are similarly based on 1990s-era knowledge. Significant advances in materials science and engineering have occurred over the past 20 years, including the emergence of computational thermodynamics as an accurate tool for rapidly assessing phase stability in structural alloys that can enable the design of improved high performance materials. Similarly, improvements in advanced manufacturing such as additive manufacturing for fabrication of geometrically complex and/or multiple-material components and friction stir welding for joining melt-sensitive alloys can enable innovative new component designs that would have been impossible 20 years ago. Several examples will be reviewed to illustrate the potential for achieving ultra-high performance and radiation resistance in new generations of structural materials for high heat flux and blanket structural applications, including new creep-resistant copper alloys and reduced activation ferritic/martensitic steels. Opportunities for utilizing advanced manufacturing in DEMO reactor components will also be summarized.

Research sponsored by the Office of Fusion Energy Sciences, U.S. Department of Energy

Eligible for student paper award?:

No

R.OP6: Safety, Operations, and Maintenance / 531

Failure Impact of crucial components on DEMO maintenance performance and mitigation attempts

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EU-DEMO should demonstrate the technical as well as the economic viability of future fusion power plants. To reach the implemented goal of a sound availability also remote maintenance has to be performed fast and highly reliable.

During regular maintenance of the DEMO tokamak fusion reactor, all blanket segments, cooling and breeder pipes, divertor cassettes and other vessel inventory have to be replaced. Even with high technical reliability of equipment and components, failures possibly occur. Some of these failures do have a minor influence on the overall maintenance progress whilst others severely impact or even prevent the further maintenance of a complete port.

There are e.g. 18 upper ports and through each of them –as crucial components - 6 bundles of cooling and breeder pipes, and 5 blanket segments have to be extracted and inserted. Before getting access to the blanket segments, all port inventory e.g. the pipes have to be removed completely. Being placed at a key position in terms of access to the in-vessel components, failures with e.g. disjoining of the pipe connections may lead to a long-term compulsory break which will reduce the availability

drastically. This is of high importance, as maintenance operations like extraction/insertion of blanket segments can exclusively take place through the dedicated port.

This contribution will emphasize the consideration of logistics aspects in terms of impact of failure on maintenance performance as well as attempts and strategies to mitigate or minimize the additional downtime caused by the failure.

To keep the downtime as short as possible, logistics as well as design aspects have to be carefully investigated and brought to an optimum. Furthermore, the design and the logistics processes have to be robust against failures. Recovery and rescue operations should be easily possible with short as possible interruptions of the maintenance process.

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

Eligible for student paper award?:

No

W.OP3: Blankets and Tritium Breeding: Solid Breeders / 532

Overview of the HCPB Research Activities in EUROfusion

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In the framework of the EUROfusion's Power Plant Physics and Technology, the Working Package Breeding Blanket aims at investigating 4 different Breeding Blanket (BB) concepts for a EU Demonstration Fusion Reactor Concept (DEMO). One of these concepts is the Helium Cooled Pebble Bed BB, which is based on the use of pebble beds of lithium ternary ceramic compounds and beryllium (or beryllides) as tritium breeder and neutron multiplier respectively, Eurofer as structural material and helium as primary coolant. The HCPB is also one of the 2 EU concepts to be tested in ITER in the frame of the Test Blanket Module (TBM) program.

This paper aims at giving an overview of the EU HCPB BB research that is being developed in KIT, in collaboration with Wigner-RCP, the BUTE-INT and CIEMAT. The paper starts giving an outline of the HCPB BB design evolution and state-of-the-art in comparison to the ITER-TBM, its basic functionalities, requirements and performances, and the associated R&D activities for the production and characterization of the ceramic breeder and Be neutron multiplier materials. It also includes a brief description of the manufacturing and testing programs aiming at a consistent realization of the HCPB BB concept through qualification of relevant mock-ups and their tests under relevant out-of-pile conditions in dedicated facilities. Similarly, an overview is given on the research in the key HCPB BB interfacing areas of Tritium Extraction and Recovery systems and their associated technology, as well as the consistent integration of the HCPB Primary Heat Transfer System into the Power Conversion System and the Energy Storage System. Moreover, an outline of the activities for

the design of a blanket attachment system as well as a description of the work performed on the structures to cope with the local high temperature with a discussion of the current standing on a credible DEMO HCPB BB system requirements.

Eligible for student paper award?:

No

T.PLN: Plenary T / 533

Status and Progress of JT-60

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JT-60SA is a highly shaped large superconducting tokamak designed to contribute to early realization of fusion energy by complementing ITER in resolving key issues such as plasma actuators (heating, current drive, perturbation coils, W-shaped divertor,

Fabrication and installation of components is steadily progressing towards start of operation. Components arrived at Naka from EU (ENEA in Italy, Germany) are preassembled with the outer inter-coil structure. The Vacuum Vessel (VV) and the thermal shield support structure coils have been installed around VV (Fig. 1) by QST with excellent accuracy of manufacturing. The support structure was also completed in the Naka site. Support structure (total 26) has been delivered from Germany. The support structure (RFX, CEA and QST) has also been implemented. The support structure is also reaching its final phase in Spain

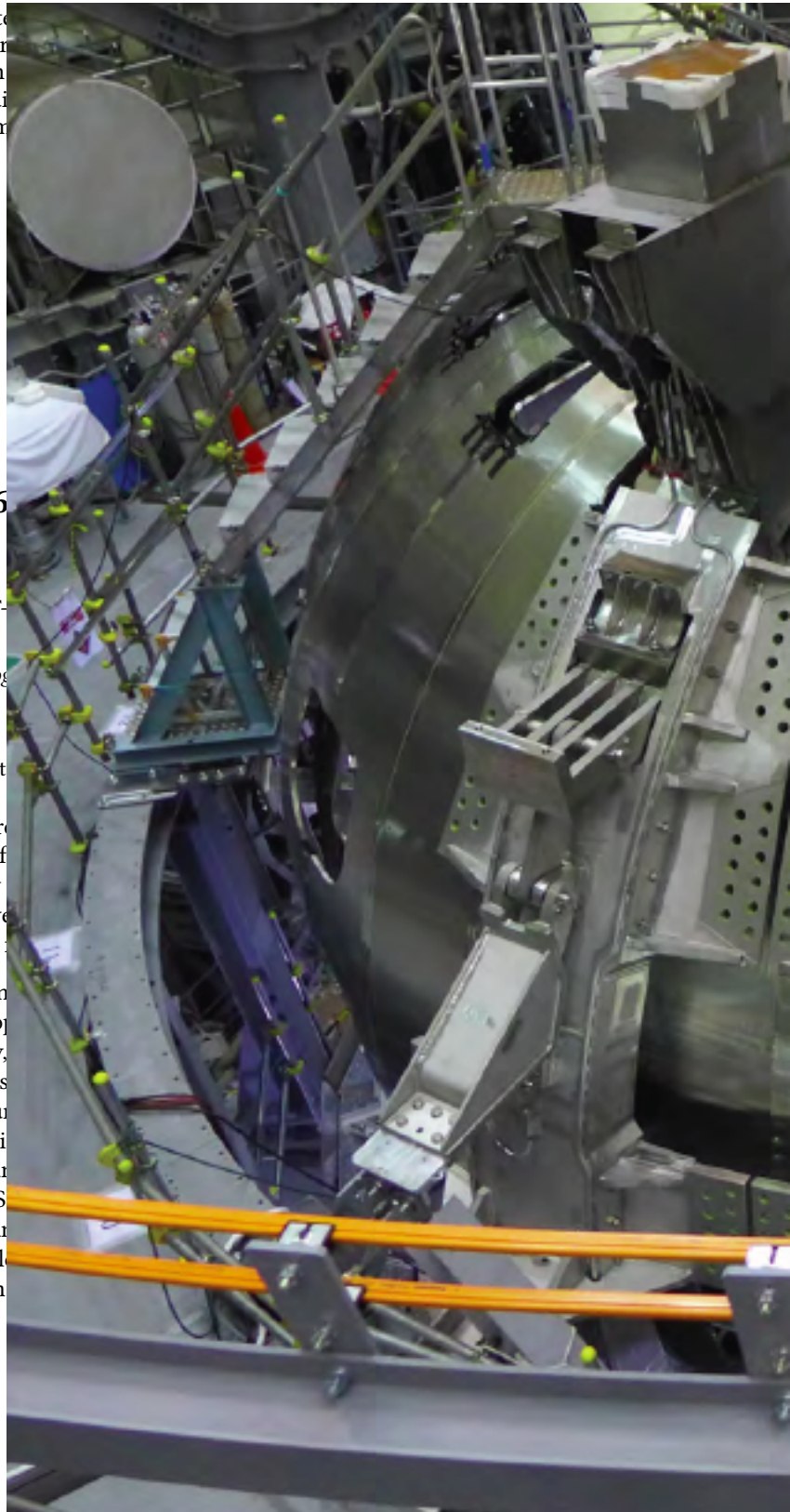


Figure 4: Fig. 1 Four TF coils have been installed around the Vacuum Vessel of JT-60SA (Mar. 2017)

Fig. 1 Four TF coils have been installed around the Vacuum Vessel of JT-60SA (Mar. 2017)

The JT-60SA Research Plan (SARP) ver. 3.3 was issued in March 2016 by 378 co-authors (JA 160 (16 institutes), EU 213 (14 countries, 30 institutes) and the STP-PT (5)). The main revision point of ver. 3.3 is the update of EU-DEMOs and JA-DEMO parameters. The revision made it clear that JT-60SA

covers wide research areas for DEMO, both pulsed and steady-state operations. The fifth JT-60SA Research Coordination Meeting (RCM) was held at QST Naka in May 2016. Contribution of JT-60SA to ITER was emphasized in relation to the expected delay of ITER. The physics R&D priorities in JT-60SA fulfilling ITER needs were suggested by the ITER Organization and discussed by all of the participants. Critical issues, such as disruption mitigation, L-H threshold power, ELM mitigation, diagnostic R&D, should be tested in JT-60SA. It became a consensus of the JT-60SA research unit to modify the basic strategy of the Integrated Research Phase II (~2030) of JT-60SA so as to start with full coverage of Tungsten divertor and Tungsten first wall and accompany the initial heating experiments of ITER.

Eligible for student paper award?:

No

W.POS: Poster Session W - Board: 103 / 534

A New Parallel IGBT Current Sharing Control for Tokamak Vertical Stabilization Current Supply System

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Tokamak reactors used in fusion plants usually require a high current supply. Different sets of coils are installed in order to confine the plasma current within the vessel, one of which is the vertical stabilization (VS) coil. The current going through the coil will produce a magnetic field, which can control the position of plasma. For example, the VS coil in ITER requires a periodic pulse current peaking at 80 kA within 0.2 seconds followed by smaller pulses of 20 kA at 10 or 6 Hz for a few seconds [1]. However, the current available power electronic devices cannot reach such a high current level without paralleling the device itself or paralleling power converters. A conceptual vertical stabilization power supply design consisting of 40 full-bridge converters in parallel was proposed in [2]. However, parallel operations of IGBTs, power devices or converters need to be controlled to eliminate/reduce the current mismatches caused by differences in device/circuit parameters such as gate resistance, stray inductance, input and output capacitances or turn-on/off signal delay. Otherwise, these parameter mismatch could lead to unbalanced current sharing and further derate or even damage the device. In this paper, current sharing issue and considerations are investigated with the vertical stabilization coil power supply topology proposed in [2] as a case study.

Firstly, the parameter mismatches will be modeled into both a detailed circuit simulation model and a system analytical model using differential equations to observe and analyze the effects of the parameter mismatch on current sharing. Current mismatches in both transient and steady states are investigated. Mismatches in transient state may cause a certain device or converter to withstand a much higher current than rated by being turned on earlier than other paralleled units. The current spikes during transient state could cause device damage immediately. For mismatches during steady state, the current level is lower than that from transient mismatch, however, a prolonged higher current than designed could cause excessive heat and faster component degradation. To suppress the current mismatch, a novel control method for the paralleled full-bridge converters was proposed where the amount of current unbalance is indicated by voltage across an inserted inductor. The voltage measurement will be used to adjust the gate emitter voltage of the power device. Due to the intrinsic dependence of emitter current to the gate voltage of IGBTs, the current rising rate during transient state and current level during steady state can then be effectively controlled. This self balancing control does not require additional current reference generation, high

end measurement devices, or complicated computation procedure, compared to most of the existing technologies. A MATLAB/Simulink simulation model was established to demonstrate that the proposed current balancing control algorithm could effectively suppress the current unbalance caused by parameter mismatches among paralleled modules. To further verify the design, real-time simulation was done with OP4510 platform.

[1] L. Yue, I. Lee, and X. Yao, "Tokamak vertical stability coil power supply based on modular multilevel converter," in 2016 IEEE Power Modulator and High Voltage Conference (IPMHVC), 2016, pp. 1–5, in press.

[2] I. Song, H. Liu, J. Li, G. Gao, E. Daly, and J. Tao, "Conceptual design of iter in-vessel vertical stabilization coil power supply system," IEEE Transaction on Applied Superconductivity, vol. 24, no. 3, 2013.

Eligible for student paper award?:

Yes

T.PLN: Plenary T / 535

Announcements

W.PLN: Plenary W / 536

Announcements

W.PLN: Plenary W / 537

Overview of US ITER Domestic Agency progress

Authors: Graeme Murdoch¹; Brad Nelson¹

¹ *Oak Ridge National Laboratory*

US ITER continues to make strong progress in its contributions to the ITER project, with ~25% (as of Feb. 2017) of First Plasma deliveries completed. In 2017, the US completed its toroidal field coil conductor contributions and will also complete delivery of steady state electrical system hardware. The central solenoid modules are now in fabrication at General Atomics in Poway, California, while support structures for the solenoid are in fabrication across the US and Europe. The US Department of Energy recently approved the "performance baseline" of the US ITER First Plasma project, which is a strong validation of US project performance. Other US contributions are continuing in design and fabrication. The US scope includes the tokamak cooling water system, ion and electron cyclotron heating systems, fueling and disruption mitigation systems, vacuum piping and roughing pumps, plasma diagnostic systems, tritium processing and Instrumentation & Controls. Progress of the US ITER Domestic agency will be presented including utilized technologies and lessons learned.

Eligible for student paper award?:

No

R.PLN: Plenary R / 538

Announcements

R.OP3: Tritium Extraction and Control / 540

Tritium extraction from HCLL/WCLL/DCLL PbLi BBs of DEMO and HCLL TBS of ITER

Author: Marco Utili¹**Co-authors:** A. Tincani¹; M. Tarantino¹¹ *ENEA FSN-ING C.R. Brasimone***Corresponding Author:** marco.utili@enea.it

The Tritium balance in DEMO Reactor is a key factor for the successful of the production of energy from Thermonuclear Fusion Reactor. Three of the four Breeder Blankets (BB) concepts candidates for DEMO used the eutectic Pb-16Li enriched at 90% in ⁶Li as breeder, the WCLL (Water Cooled Lithium Lead), DCLL (Dual Coolant Lithium Lead) and HCLL (Helium Cooled Lithium Lead) BBs, therefore the design and characterization of the Tritium Extraction and Removal System (TER) from PbLi with high efficiency is a critical issue in the European Roadmap. In ITER Research Reactor a PbLi BB concept will be qualified, the HCLL BB. A HCLL Test Blanket Module, PbLi loop, instrumentations and auxiliary systems will be characterized with the support of European infrastructures. However, the Tritium Extraction Unit from PbLi (TEU) selected and designed for ITER, is based on Gas Liquid Contactor technology, a reliable technologies but with less than 40% efficiency. Instead, the TER candidates technologies of DEMO BBs, the PAV (Permeator Against Vacuum) and VST (Vacuum Sieve Tray), will not qualified in ITER because these systems will not be fully mature by the start of the Reactor. PAV and VST can theoretically achieve efficiency above 80%.

The present works aims to analyse the technologies candidates for ITER and DEMO reactors, describe and compare TEU and TER design for each concept of BB and the integration of TER in DEMO tokamak building taking into account two design requirements: self-sufficient sustainable of fusion nuclear reactor and safety requirements.

Eligible for student paper award?:

No

M.PLN: Plenary M / 541

Opening Remarks

Corresponding Author: hneilson@pppl.gov**R.OP1: Diagnostics and Instrumentation II / 542**

Managing ITER diagnostics and port plug engineering project risks

Author: Russell Feder¹

Co-authors: Frank Hoffmann¹; Ankita Jariwala²; Mark Smith²; Roger Reichle³; Vincent Martin⁴

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Risk management is a key part of any successful project. Projects like ITER with high technical challenges and complex integration vitally depend on risk management. This paper will describe the organization and management of risks within the ITER diagnostics and port plug engineering project design phase. A typical risk impacts design-phase cost and schedule due to the need for specific mitigation actions and additional design-by-analysis or design-by-prototype iterations. The paper will describe the management of five categories of design-phase engineering risks in the context of the Upper Wide Angle View (UWAVs) Visible and Infrared Camera diagnostic project. UWAVs is the largest US ITER diagnostic system and serves critical machine protection roles. The system provides a comprehensive view of the divertor and outer blanket wall for hot spot detection as well as visible inspection of these critical structures. The first type of design-phase risks are associated with the optimization and integration of the in-vacuum diagnostic port structures including balancing weight with shielding effectiveness, structural performance and diagnostic performance. As UWAVs is deployed in five upper port plugs there are some space-claim clashes to deal with despite strong IO CAD model controls. One mitigation approach for UWAVs is to share light collecting optics with another diagnostic. UWAVs proposal to share optics with the RFDA H-Alpha project will be discussed in this talk. The second risk category deals with the ITER operating environment and the need for radiation and magnetically hardened components. This is a critical risk category for UWAVs because the visible and IR cameras sit in high radiation areas. The UWAVs project approach to this risk will be discussed in this talk. Risks associated with design strategy and assumptions make up the third grouping. Strategy and assumptions are also the “risk” category that may lead to positive outcomes with cost savings and schedule acceleration. There are several examples of strategy from the UWAVs project. The major design strategy for UWAVs, also related to integration, is the decision by the US ITER project to have one UWAVs design for all five upper port deployments. The pros and cons of this strategy will be discussed in this talk. The fourth group deals with technology development through R&D and prototype activities. This is a key category because aspects of design are strongly linked to choices of technology. R&D and prototype work also helps to mitigate performance risks once the system is deployed. In the UWAVs project risks inherent in design strategy and technology development are partly associated with first mirror cleaning. The project has based the in-vacuum design on RF driven sputtering technology. A comprehensive mirror cleaning R&D and prototype program is underway in the US and several other DAs in order to handle this risk of design strategy based on new technology. Lastly risks associated with changing interfaces or requirements will also be discussed.

Eligible for student paper award?:

S.PLN: Plenary S / 543

Progress of Fusion Technology at SWIP toward Reactor

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China has made its fusion energy development strategies with a route from ITER over through a test reactor towards the future DEMO reactor. As one of the institute for fusion energy development in China, SWIP is focusing on investigation of plasma physics and magnetically confined nuclear fusion by operation the HL-2A Tokamak. In addition, SWIP has been engaged in the development of fusion technologies in recent 10 years, majoring in fusion reactor materials, tritium breeding blanket and divertor by joining the ITER programme and the domestic fusion energy

development projects. SWIP is leading R&Ds on the ITER TBM in China and has developed the technologies for ITER first wall, shielding blanket modules, magnet support and so on through a number of ITER procurement arrangements. Two research centers were established in SWIP for developing the technologies for blanket and divertor for fusion reactors. A lot of facilities have been built and put into use for manufacturing and testing the ITER in-vessel components, developing advanced materials and the material joining technologies, such as the 1400°C/200 MPa hot iso-static pressing machine for thermal diffusion bonding plasma facing armor materials to heat sink or steel structures, the 400kW EB facility for high heat flux test of FW and divertor component, a 8 MPa/300-500°C Helium testing loop for studying the hydraulic performance of the Chinese HCCB TBM. CLF-1 RAFM steels in 5 tons scale has been made with acceptable physical and mechanical properties for ITER TBM. Small scale TBM mock-ups have been manufactured along with extensive investigation of welding technologies, including TIG welding, Laser and EB welding, HIP bonding, etc. Production of Be pebbles as neutron multiplier and Li_4SiO_4 tritium breeder have been demonstrated in laboratory and are scaled up to 10 kg scale for future application. Beryllides are made in laboratory as promising candidates for advanced neutron multipliers. The bonding of Be and W tiles to CuCrZr alloy heat sink has been successfully performed by HIP joining and brazing to develop the FW and divertor technologies. Be/CuCrZr Mock-ups and an EHF FW semi-prototype were made, which are tolerable for 4.7 MW/m^2 for the ITER designed number of cycles. W/CuCrZr divertor mock-ups survived successfully from 10 MW/m^2 power density for 1000 cycles and 15 MW/m^2 for 300 cycles. Recently the technologies for a small unit of advanced helium cooled divertor were developed, for which a couple of $\text{W-Y}_2\text{O}_3$ and alloys were made with expanding application temperature window. A 1-10 eV ($10^{22} \sim 10^{23} \text{ m}^{-2} \text{ s}^{-1}$ in flux) linear plasma device is under construction for plasma-material interaction studies with the synergistic effects of “plasma-heat-displacement damage” on PFMs. For better high-temperature performance, alternative structural materials such as ODS RAFM steel and carbide dispersion strengthened (CDS) vanadium alloys have been studied. This paper will show the status and highlight the progress in past years in SWIP for the fusion technologies for ITER and the reactor beyond in China.

Eligible for student paper award?:

No

M.OP3: Next Step Devices, DEMO, Power Plants / 544

Status of the US Virtual Laboratory for Technology

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The Department of Energy Fusion Energy Sciences (FES) program created the Virtual Laboratory for Technology (VLT) in 1998 with the goal of establishing a single entity with central leadership that would connect all aspects of the FES Technology Program. Since its inception, the VLT has been successful in conducting high quality research in support of the fusion energy sciences program mission as well as representing the technology program within the fusion community. The VLT represents the diverse activities of US Universities, Laboratories, and Industry involved in fusion technology research and development, and is organized into technical program elements that span the spectrum of technologies required to carry out its mission.

While technical work at each of the member institutions has continued, the VLT has been reinvigorated over the last year following a roughly two-year hiatus. An overview of the current activities, new activities that have recently been initiated, and discussions of plans will be presented.

Eligible for student paper award?:

No

R.OP6: Safety, Operations, and Maintenance / 545

Adjourn

M.PLN: Plenary M / 546

Opening Remarks

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M.PLN: Plenary M / 547

Welcome Message

M.PLN: Plenary M / 548

Conference Logistics

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M.PLN: Plenary M / 549

Technical Program Overview

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W.POS: Poster Session W - Board: 79 / 550

Study of pumping speed of Activated Carbon based Cryosorption Pump

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The search for alternative means of power source to meet the need of ever increase power demand is eternal, given the present global scenario a clean and sustainable form of energy with sufficiency

to meet the demand is being investigated. One such form of energy is Nuclear fusion reaction. On the onus of having a harsh condition and a requirement of high pumping speed Cryosorption pumps have emerged as an only alternative for the removal of Hydrogen and Helium in fusion reaction systems. The development of such pumping mechanism requires the appropriate Activated Carbons (ACs) and suitable adhesives to bind them to the metallic panels with liquid helium (LHe) flow channels. However, their performance evaluation will require huge quantities of LHe.

Alternatively, these cryopumps can be fabricated with small size panels adhered with ACs and cooled by a commercial cryocooler. The alternative for avoiding the use of LHe has lead us to the development of a cryopump using a commercial cryocooler, with 1.5W at 4.2 K, combined with small size AC panel mounted on 2nd stage, while the 1st stage will be acting as radiation shield. Under no load, the cryopump reaches the ultimate pressure of 2.1×10^{-7} mbar. The pump is fabricated using panels with different indigenously developed ACs such as granules, pellets, ACF-FK2 and activated carbon of knitted IPR cloth. In this report, we present the experimental results of pumping speeds for various gases such as nitrogen, hydrogen, argon and helium using the procedures prescribed by the American Vacuum Society (AVS). These studies will enable us to arrive at the right ACs and adhesives required for the development of large scale cryosorption pumps with liquid helium flow.

Eligible for student paper award?:

Yes

TOP1: Power Supply Systems / 551

Adjourn