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INVITED AND ORAL SESSIONS
MONDAY 16TH SEPTEMBER
09:45 – 10:30
Opening Ceremony | Auditorium

I1.1 Status of the ITER project (ITER DG)

Motojima, Osamu

PLENARY SESSION 1
11:00 – 13:00
Auditorium

I1.2 A Road Map to the realization of Fusion Energy

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The human population is expected to double by 2100 putting a strong demand on energy production. Due to CO_2 production limits and availability of fuel resources by 2100 it is clear that the demand of energy for future generations must and can only be supplied in a mix of unconventional renewal and reusable energy sources. Among the emerging technologies for energy production fusion can play a crucial role as a long lasting, environmentally friendly high density energy source.

To provide fusion in the time pace set by the boundary conditions of population growth, environment aggression, fuel fossil availability and technology developments an ambitious, yet realistic roadmap towards the realization of a demonstration of a thermonuclear fusion power plant (DEMO) for electricity production by 2050[1] has been established.

The roadmap has been developed within a goal-oriented approach articulated in eight different Missions. The critical aspects for reactor application, the risks and risk mitigation strategies, the level of readiness now and after ITER and the gaps in the programme have been examined with involvement of experts from the ITER International Organization, Fusion for Energy, EFDA Close Support Unites and EFDA Associates. High-level work packages for the roadmap implementation have been prepared and the resources evaluated.

ITER is the key facility in the roadmap**.** ITER will break new ground in fusion science and the European laboratories shallfocus their effort on its exploitation.

1 F. Romanelli, P. Barabaschi, D. Borba, G. Federici, L. Horton, R. Neu, D. Stork, H. Zohm *A roadmap to the realization of Fusion Energy* (2012)

Additional challenges that are key to be addressed in line with the design of DEMO must be assessed. A reliable solution to the problem of heat exhaust is probably the main challenge towards the realisation of magnetic confinement fusion. A dedicated neutron source for irradiation studies with a fusion neutron spectrum is needed before the DEMO design can be finalised. The tritium self-sufficiency technology should be strengthened beyond ITER. DEMO blanket selection will face new constraints on coolant and breeder arising from the choice of an efficient Balance of Plant.

For such a long term development programme two additional ingredients are essential to provide the required commitment for its technologic success: industry and international cooperation.

Industry must be involved early in the DEMO definition and design. The evolution of the programme requires that industry progressively shifts its role from that of provider of high-tech components to that of driver of fusion development. Industry must be able to take full responsibility for the commercial fusion powerplant after successful DEMO operation.

Europe should seek all the opportunities for international collaborations. Some of the ITER parties have a very aggressive programme in fusion and Europe can clearly benefit by the participation in the design, construction and operation of their facilities. Collaboration that can give further advantages on the time scale considered here.

I.1.3 Overview of Activities and Strategy in Japan on Fusion Technology for DEMO

Horiike, H; Hashizume, H.; Konishi, S.; Nishitani,T.; Okano, K.; Ogawa, Y.; Shimizu, K.; Mori, S. and Tobita, K.

For early realization of the nuclear fusion energy, the broader approach projects are carried out under the collaboration of JA and EU in parallel with ITER. Japanese road map to the DEMO reactor was studied, and many research subjects which were in advance necessary to judge the validity of its design specifications and of the engineering design subjects for DEMO were found to be left behind of prospects of ITER and the broader approach project.

Our road map to DEMO is devided into two phases of the conceptual design, and of engineering design with R&D. These project works are required to be led by newly organized so-called strategic design core team.

The design and R&D works include those on super conducting coils, blanket and tritium breeding, divertor development, theory and numerical simulation, core plasma study, fueling and fuel systems, tritium safety technology, reactor materials and design standards, and current drivers, in addition to the human resource development. These subjects are extracted and proposed to the community to include these in the course of DEMO design and the R&D scenario development, in parallel with the main projects of ITER and broader approach. In the presentation the general pictures of these route surveillance will be presented and discussed.

I.1.4 K-DEMO Design, R&D Plan, and International Collaboration

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In late 1995, having launched the KSTAR Project as the first project of the "National Fusion Energy Development Plan", Korea promoted "Mid-entry Strategy" to make a leap forward to a full-scale Fusion Energy Research and Development Program. Moreover, adopting new technologies such as Nb3Sn superconducting magnets in full-scale, succeeded in developing a new generation of steady-state fusion research facility. After the successful construction of KSTAR, Korea joined the ITER as an initial signatory partner with six other Member States.

With the "ITER construction" well underway, there is a growing anticipation among ITER Members for an earlier realization of fusion energy with a practical demonstration of electricity generation on a power plant scale machine, a Demonstration Fusion Power Plant (DEMO). In Korea, preparation for the last leg of "Mid-entry Strategy", namely a Korean DEMO (K-DEMO) design and R&D planning activity, was initiated in 2012. The aim is to launch a new national program in 2014, and to bridge technological gaps that exist between ITER plasma and nuclear regime and that of DEMO.

In design and specifications, K-DEMO should be just a small step away from a commercial plant in terms of technology and performance with adopted key metric as Reliability, Availability, Maintainability, Efficiency, and Safety (RAMES). Also, K-DEMO is planned to be a two-stage project. The first stage, called K-DEMO 1, will develop and test components for the second stage, K-DEMO 2, to produce fusion energy and generate electricity. We will present the results of preliminary conceptual design study of K-DEMO and the implementation plan for core technology R&D based on a gap study with priority.

14:30 – 16:00
PARALLEL M1 (Topic B: Blankets- Special TBMs)
Auditorium

O1A.1 The European Test Blanket Module Systems: Status of design, technologies development and integration in ITER

Poitevin, Y.[1]; Ricapito, I.[1]; Zmitko, M.[1]; Panayotov, D.[1]; Calderoni, P. [1]; Galabert, J. [1]; Vallory, J.[1]; Giancarli, L. [2]; Tavassoli F. [3]; Thomas, N. [4]; De Dinechin, G.[3]; Bucci, Ph. [5]; Aiello, A. [6,10]; Rey, J.[7,10]; Rueda, F.[8]; Ibarra, A.[9]

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Europe has developed two reference tritium Breeder Blankets concepts that will be tested in ITER under the form of Test Blanket Modules (TBMs): i) the Helium-Cooled Lithium-Lead (HCLL) which uses the liquid Pb-16Li as both breeder and neutron multiplier, ii) the Helium-Cooled Pebble-Bed (HCPB) with lithiated ceramic pebbles as breeder and beryllium pebbles as neutron multiplier. Both concepts are using the EUROFER reduced activation ferritic-martensitic (RAFM) steel as structural material and pressurized Helium technology for heat extraction (8 MPa, 300-500ºC).

In a first part, the paper addresses a key challenge of the European TBMs project in ITER consisting in designing, constructing and licensing a nuclear component – the TBM – that is featuring recently developed materials and technologies (e.g. EUROFER, tritium breeder materials, welding technologies, etc.). The European strategy is consisting in capitalizing the related developments in the nuclear design & construction code RCC-MRx in view of both the licensing of TBMs in ITER and the future construction of DEMO tritium breeder blanket. To this regard, Europe has achieved a first key milestone with the introduction of the EUROFER material in the RCC-MRx 2012

edition (design rules are first under probationary phase). Also, non-conventional fabrication procedures for the TBM box are presently under development/qualification through collaboration between European Fusion Laboratories and Industry for the establishment of standardized welding procedure specifications. Their suitability is being demonstrated through medium to full-size mock-ups; fabrication procedures will then be qualified according to European standards before construction of the first TBMs.

In a second part, the paper reviews the recent developments and studies related to the expected performance of the TBM Systems testing in ITER and prediction capability; in particular, recent development in the area of tritium cycle modeling is summarized. The TBMs test strategy foresees to develop and install successively in ITER various TBMs specifically designed and instrumented to optimize the scope of achievable tests with respect to the different ITER operational phases. The paper analyzes the TBM Systems test objectives for the presently envisaged ITER research plan as well as the specific development needs for TBM Systems instrumentation and the use of ITER diagnostics for maximization of the TBM program outcome.

In view of licensing TBM Systems in ITER, Europe has also recently issued Preliminary Safety Reports aimed at demonstrating that TBM Systems fulfill regulatory and ITER safety requirements. The paper reviews the key safety provisions and outcome that are specific to the European TBM Systems.

At last, the potential impact on the TBM Program of possible mitigation measures to limit the ITER toroidal field ripple induced by TBM ferromagnetic structures is discussed.

O1A.2 Design and R&D Progress of Korean HCCR TBM

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Korea is supposed to develop and test Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM) in ITER. The HCCR TBM is composed of four sub-modules considering fabricability and the transfer of irradiated TBM for post irradiation examination. Each sub-module has seven layer breeding zone, including three neutron multiplier layers packed with beryllium pebbles, three lithium ceramic pebble packed tritium breeder layers and a reflector layer packed with silicon-carbide (SiC) coated graphite pebbles to reduce the volume of beryllium. Reduced activation ferritic-matensitic (RAFM) steel being developed in Korea will be used as a structural material. Based on this configuration, neutronics and electromagnetic calculation have been performed, and their results are applied for the conceptual design of HCCR TBM with cooling scheme and coolant manifold design considering manufacturing feasibility. Also, the design and safety analysis of HCCR Test Blanket System (TBS) has been performed using integrated design tools modifying nuclear system codes for helium coolant and tritium behavior evaluation.

Currently, R&Ds in several areas have been performed to develop HCCR TBS. In order to estimate the manufacturability, thermo-hydraulic and high heat flux performance of four sub-module TBM, a half scale mockup of a sub-module is fabricated and tested in the recently established high heat flux test and helium cooling facilities, and their results are used for the validation of design tool. Also, HCCR TBM materials have been developed. Several RAFM steels have been designed and their out-of-pile performance is being evaluated. And, the possibilities of mass production of lithium ceramic and graphite pebbles applying a CVR-SiC coating method are investigated. The thermo-mechanical properties of these pebble beds are assessed. The hydrogen permeation characteristics of stainless steel and RAFM steels are evaluated using Hydrogen PERmeation (HYPER) facility. Recent design and R&D progresses on these areas are to be introduced in this paper.

O1A.3 Status of Indian LLCB TBM program and R&D activities

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India is currently engaged in the development of Lead-Lithium cooled Ceramic Breeder (LLCB) blanket system, which is the primary option of Indian Test Blanket Module (TBM) program towards the realization of DEMO reactor. The LLCB TBM will be tested from the first phase of ITER operation (H-H phase) in one-half of the ITER port no-2. The LLCB blanket concept consists of lithium titanate as ceramic breeder (CB) material in the form of packed pebble beds. The FW structural material is ferritic steel cooled by high-pressure helium gas and Lead-Lithium eutectic (Pb-Li) flowing separately around the ceramic breeder pebble bed to extract the volumetric heat from the CB zones. Recently, the LLCB blanket design has been optimized to parallel flow configuration based on the neutronic as well as thermal hydraulic analysis results. The design of the shield block is under progress. The mechanical design of the TBM structure integrated with the shield block and ITER port plug has been initiated. Significant progress has been made in the process design of LLCB TBM auxiliary systems such as helium cooling systems (HCS), Lead-Lithium cooling system (LLCS) and helium purge system (HPS).

The Indian TBM R&D activities are primarily focused on (i) Development and characterization of blanket materials such as structural (IN-RAFMS), Tritium breeding materials (Pb–Li, and Li_2TiO_3) (ii) Development of technologies for critical components such as circulators, pumps, heat exchangers, diagnostics etc. for Lead-Lithium cooling systems, Helium Cooling Systems and Tritium Extraction Systems, (iii) Development and testing of manufacturing technologies for TBM system. Lead-Lithium technologies development activities are focused on liquid metal loop developments for MHD experiments, corrosion experiments, diagnostics calibration and operational experience with critical loop components. Work is also in progress to develop numerical codes and their benchmarking with the available experimental results.

This paper will describe the present status of LLCB TBM design and various R&D activities.

14:30 – 16:00
PARALLEL M2 (Topic D: Materials)
Room 6

O1B.1 Fusion Materials Science and Technology Research Opportunities now and during the ITER Era

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Achievement of practical fusion energy will require resolution of numerous materials science and engineering issues largely stemming from the extreme operating environment of a fusion reactor. This presentation summarizes an evaluation of fusion materials science and technology research opportunities for the next ten years. Three overarching grand challenges were evaluated that comprise fusion nuclear science: Taming the plasma-materials interface, Conquering nuclear degradation of materials and structures, and Harnessing fusion power (tritium science, chamber technology and power extraction). The scientific challenges associated with these three fusion nuclear science themes are extraordinary. For plasma-materials interactions, the material surfaces directly facing the plasma are exposed to continual energetic bombardment of plasma particles that both exhaust heat and “recycle” the hydrogen fuel; for example a surface atom may be

removed and redeposited over a billion times in a single year. Neutron radiation damage to materials and structures involve atomic- and meso-scale physical processes that span more than 20 orders of magnitude in time scale and over 8 orders of magnitude in length scale. The magnetohydrodynamic interactions of flowing liquid metal coolants in spatio-temporally varying magnetic fields leads to highly non-linear 3D fluid physics, which can exceed viscous and inertial forces by five or more orders of magnitude – dominating the flow behavior and heat transfer and thereby controlling the ultimate operating temperature, pressure, stress fields, and transport properties of the in-vessel systems. Finally, tritium must be handled at unprecedented flow rates of many kilograms per day, with efficient processing over a wide range of temperatures, pressures and material conditions (where vastly different chemical mechanisms are operative), while observing stringent accountancy and environmental release constraints.

Several high-priority near-term potential research activities to address these fusion nuclear science challenges will be summarized. General recommendations include: 1) Research should be preferentially focused on the most technologically advanced options (i.e., options that have been developed at least through the single-effects concept exploration stage, Technology Readiness Levels >3), 2) Significant near-term progress can be achieved by modifying existing facilities and/or moderate investment in new medium-scale facilities, and 3)Computational modeling for fusion nuclear sciences is generally not yet sufficiently robust to enable truly predictive results to be obtained, but large reductions in risk, cost and schedule can be achieved by careful integration of experiment and modeling.

O1B.2 Modeling Plasma Surface Interactions in Tungsten through High-Performance Computing

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The plasma facing components, first wall and blanket systems of future tokamak-based fusion power plants arguably represent the single greatest materials engineering challenge of all time. Indeed, the United States National Academy of Engineering has recently ranked the quest for fusion as one of the top grand challenges for engineering in the 21^{st} Century. These challenges are even more pronounced by the lack of experimental testing facilities that replicate the extreme operating environment involving simultaneous high heat and particle fluxes, large time varying stresses, corrosive chemical environments, and large fluxes of 14-MeV peaked fusion neutrons. 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 materials performance and degradation in the fusion energy environment. This presentation will describe the challenges associated with modeling the performance of plasma facing component and structural materials in a fusion materials environment, the opportunities to utilize high performance computing and then focus on an example of recent progress to investigate the dramatic surface evolution of tungsten exposed to low-energy He and H plasmas. More specifically, multiscale modeling results will be presented to identify the mechanisms of tungsten surface morphology changes when exposed to 100 eV He plasma conditions as a function of temperature and initial tungsten microstructure. The results demonstrate that during the bubble formation process, He clusters create self-interstitial defect clusters in W by a trap mutation process, followed by the migration of these defects to the surface that leads to the formation of layers of adatom islands on the tungsten surface. As the helium clusters grow into nanometer sized bubbles, their proximity to the surface and extremely high gas pressures leads them to rupture the surface thus enabling helium release. Helium bubble bursting induces additional surface damage and tungsten mass loss which varies depending on the nature of the surface.

O1B.3 Neutron Irradiation Effects on Tungsten Materials.

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Since tungsten (W) has a high melting point and a high sputtering resistance to energetic particles, it is considered to be a candidate for the plasma-facing component (PFC) materials of magnetic confinement fusion reactors such as the first wall of a blanket and a diverter plate. During fusion reactor operation, as a result of high-energy neutron exposure, not only displacement damage but also transmutation elements are produced. The defects and transmuted elements cause hardening, embrittlement, decreasing thermal conductivity etc. These irradiation effects of material properties strongly depend on irradiation conditions such as irradiation temperature, irradiated fluence, neutron energy and also depend on material source, interstitial impurities and fabrication processes. This study will present a brief summary of the current knowledge-base of irradiation behavior of tungsten and results of our data on neutron irradiation behavior of various type tungsten alloys below 1dpa, and will show predictions of material performance change of tungsten materials under fusion reactor irradiation environments.

14:30 – 16:00
PARALLEL M3- (Topic E: Exvessel)
Room 5

O1C.1 Challenging issues in the design and manufacturing of the European sectors of the ITER Vacuum Vessel

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Fusion For Energy (F4E), the European Domestic Agency for the ITER project, has to supply seven sectors as part of the European contribution to the project. F4E signed the Procurement Agreement with ITER Organization (IO) in 2009. After a call for tender in 2010, the contract for the manufacturing of seven sectors was placed in October 2010 to a consortium of three Italian companies, Ansaldo, Mangiarotti and Walter Tosto (AMW). The first sector in the manufacturing route is Sector 5. This paper will cover the status of the engineering activities, design, procurement and preparation to begin the manufacturing in 2013. Also will be presented the statutory and regulatory requirements of the French Nuclear Safety regulator and the status of the relevant R&D mock-ups to demonstrate manufacturing feasibility control of distortions (using predictions with analysis and algorithms to change in real time the manufacturing route in order to correct such distortions, inspectability and metrology). Another important aspect at this stage of the manufacturing is qualification of activities like welding, Non-destructive Examination and Hot Forming.

This paper describes the status of the activities currently in process in order to meet with the challenging design, schedule and high quality requirements of the project.

Keywords: ITER Vacuum Vessel, Manufacturing, Welding, NDE Inspection

O1C.2 Pre-conceptual studies and R&D for DEMO Superconducting Magnets

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The DEMO plant will demonstrate by mid century the feasibility of electric power generation by nuclear fusion. Since 2011, conceptual design studies are coordinated by the European Commission, with the aim of identify requirement, propose design approaches and start R&D for the magnet system of DEMO. The input and generic boundary conditions are given by the System Codes: the major radius of the tokamak is 9 m. The proposed operating current at 13.5T peak field is 85kA, placing the DEMO TF conductor at substantially higher performance compared to ITER TF (68kA / 11.5T). The innovative winding layout is a graded, layer wound with Nb_3Sn / NbTi hybridization for both conductor designs, aiming at minimizing the size and the cost of the superconductor: one of the conductors is a "wind & react" CICC with reduced void fraction and rectangular shape. The other conductor is a "react and wind" flat cable with copper segregation and thick conduit assembled by longitudinal weld. The conductor design were first drafted in 2012 and updated in 2013 based on a first round of assessments, which includes electromagnetic, thermal-hydraulic and mechanical analysis. The manufacture of full size prototypes is planned in 2014.

The technical requirement of the DEMO superconducting magnets is highlighted in comparison to ITER and other fusion devices. The large size of the DEMO tokamak is the main challenge for the demonstration of the feasibility of power generation by fusion. Together with the technical issues, the cost of the superconducting magnets will be eventually the crucial aspect to promote the establishment of nuclear fusion as a primary energy source in the coming centuries.

O1C.3 Challenging Issues in the Manufacturing of the ITER cryostat

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One of the ITER-India's commitments to ITER is the Cryostat. ITER Cryostat is a large vacuum vessel (~29m dia. and ~29 m height), which will be made up of Stainless steel and will have a total mass of more than 3500 Tons. The thickness of vessel wall varies from 50mm to 180mm. It is one of the heaviest fully welded 'steel vacuum chamber' in the world. Cryostat will provide vacuum tight environment for ITER along with absorption of thermal loads and support to Vacuum Vessel and super conducting magnet systems.

The reference Code of manufacturing/fabrication for ITER Cryostat will be ASME Section VIII Division-II along additional vacuum requirements as ITER Vacuum Handbook and strigent manufacting & assembly tolerances required for ITER Machine.

Its manufacturing will be done in small segments in India consistent with the transportation constraints in France; sub-assembly of four major sections of the Cryostat from segments will be done at ITER site in a Temporary Workshop (TW) and final assembly will be done inside the bio-shield enclosure (tokamak pit) in a phased manner, where it will be finally installed. We need to ship cryostat in smaller segments to port at Marseille and from thereon it will be transported by road to the ITER site at Cadarache, France.

A Temporary Workshop having a size of about 50mx120m and a height of 30m will be erected along with a 200T Goliath Crane at ITER Site for sub-assembly of Cryostat major sections. Special toolings and fixures will be needed for the assembly of Cryostat sections in the constrained working area of tokamak pit.

The talk will focus on constraints and challenges forseen in manufacturing design, welding-inspection & testing, tolerance control, sub-assembly and Pit-installation of Cryostat. It also discusses processes for assembly of Cryostat sections in tokamak building and manufacturing procedure design in terms of Quality Assurance and application of Codes & Standards, Safety related issues etc.

16:30 – 18:00
PARALLEL M4 (Topic B: Blankets – Special TBMs)
Auditorium

O2A.1 New Progress on Design and R&D for Solid Breeder Test Blanket Module in China

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Southwestern Institute of Physics, Chengdu, China

ITER will be used to test tritium breeding module concepts, which will lead to the design of DEMO fusion reactor design demonstrating tritium self-sufficiency and the extraction of high grade heat for electricity production.

The helium-cooled ceramic breeder (HCCB) test blanket module (TBM) is the primary option of the Chinese TBM program. The preliminary conceptual design of CN HCCB TBM has completed. Basic characteristics and main design parameters of CN HCCB TBM are introduced briefly. The mock-up fabrication and component tests for Chinese test blanket module have being developed. Recent status on the components of CN HCCB TBM and fabrication technology development are also reported. The neutron multiplier Be pebbles, tritium breeder Li4SiO4 pebbles, and structure material CFL-1 are being prepared in laboratory scale. The fabrication of pebble bed container and experiment of breeder pebble bed will be started soon. The fabrication technology development is proceeding to R&D phase of the large scale mock-up fabrication and demonstration tests toward ITERTBM testing.

O2A.2 DCLL blanket: Status and critical R&D needs

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In the Dual-Coolant Lead-Lithium (DCLL) blanket, eutectic alloy lead-lithium (PbLi) is used as breeder/coolant in the breeding zone, while helium (He) gas is used for cooling ferritic (RAFM) structure. The key element of the concept is a flow channel insert (FCI), which serves as electrical and thermal insulator to reduce the impact of the magnetohydrodynamic (MHD) pressure drop and to decouple the temperature-limited ferritic walls from the "hot" PbLi. Using FCI allows for high PbLi exit temperature of 700°C or higher leading to high blanket thermal efficiency above 40%. Silicon carbide (SiC) is a strong option for FCI for high temperature operation of PbLi. The special features of DCLL, e.g. configuration, multiple zones/materials and operating parameters, fundamentally influence fluid flow behavior and result in extremely complex transport processes of momentum, heat and mass transfer in the presence of a strong magnetic field and volumetric heating.

Starting from 2004, several R&D studies have been launched in the US to address these and other important phenomena. First, various MHD and Heat Transfer investigations were conducted in the framework of the ITER TBM program to elucidate coupled electromagnetic – fluid flow processes in the multimaterial blanket domain composed of PbLi, RAFM and SiC. The main research objective was to characterize the effect of SiC FCI on MHD pressure drop, interfacial temperatures and heat leakages from the PbLi into the cooling He streams. In addition to conventional single-layer FCI, a double-layer FCI (nested, nFCI) was proposed and analyzed, which mitigates the thermal stress while providing sufficient thermal insulation and reducing the MHD pressure drop. In more recent studies, significant consideration is given to heat and mass transfer phenomena that include MHD induced corrosion of RAFM and transport of activated corrosion products, tritium transport, and buoyancy-driven flows. We also consider unsteady flow phenomena, including several instability types and laminar-turbulent transitions. In this context, we discuss, here, a newly constructed MHD PbLi loop at UCLA and ongoing experiments on MHD pressure drop reduction and MHD instabilities. We also review status and give examples of application of MHD/Heat & Mass Transfer computational tools such as MHD code HIMAG and mass transfer codes CATRYS and TRANSMAG.

In the paper, we also discuss critical needs, the near-future work and associated test facilities on the pathway towards the DCLL blanket that are essential to advance scientific understanding of multiple effects, and to prepare for efficient and useful testing in the full fusion environment of an ITER TBM and/or FNSF. We propose and describe requirements for near term laboratory facilities and experiments as well as *Blanket Mockup Thermomechanics/Thermofluid Test Facilities* required to understand and resolve the key technical issues.

O2A.3 R&D status on Water Cooled Ceramic Breeder Blanket Technology

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The development of a Water Cooled Ceramic Breeder (WCCB) Test Blanket Module (TBM) is being performed as one of the most important steps toward DEMO blanket in Japan. For the TBM testing and development of DEMO blanket, R&D has been performed on the module fabrication technology development, breeder and multiplier pebble fabrication technology development, tritium production rate evaluation research, as well as structure design activities.

The fabrication technology development using F82H has been performed based on the evaluation of physical properties of the structural material, F82H. The fabrication of a real scale first wall, side walls, a breeder pebble bed box and assembling of the first wall and side walls have been performed, previously. This year, the fabrication of a real scale mockup of the back wall of TBM was completed. Also the assembling of the complete box structure of the TBM mockup and planning of the pressurization testing was studied. The development of advanced breeder and multiplier pebbles for higher chemical stability was performed for future DEMO blanket application. From the view point of TBM test result evaluation and DEMO blanket performance design, the development of the blanket tritium simulation technology, investigation of the TBM neutronics measurement technology and the evaluation of tritium production and recovery test using D-T neutron in the Fusion Neutronics Source (FNS) facility has been performed. This paper overviews the recent achievements of the development of the WCCB Blanket in Japan.

16:30 – 18:00
PARALLEL M5- (Topic D: Materials)
Room 6

O2B.1 Limits and Potential of Reduced Activation Ferritic/Martensitic Steels

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Reduced activation ferritic/martensitic (RAFM) steels are now considered to be the candidates for structural applications in the fusion demonstration reactor, DEMO, because they have a sound engineering basis and rich database including irradiation database. But it is also well recognized that the severe DEMO operating conditions, especially 14MeV fusion neutron irradiation, could cause extra degradation of mechanical properties and element transmutations.14MeV fusion neutron irradiation database will not available until IFMIF irradiation will start, and this indicates that RAFM steel could be used within the irradiation range where the 14MeV fusion neutron irradiation effects could be estimated not too pronounced one, and the fission neutron irradiation data can be used as the substitutable data. On the other hand, there is a practical limit on removing undesired impurities, such as Co, Cu, Ni, and nuclear transmutation reaction could reduce the concentration of some key elements, such as W, which are essential for mechanical properties. These facts indicates that the service limit of RAFM steels for initial DEMO reactor should be defined in view of mechanical property, applicability limit of fission database, and waste management scenario.

In this paper, limits and potential of RAFM steel are discussed based on actual achievement of F82H, Japanese RAFM steel, and DEMO design activity in Japan.

Keywords: Reduced activation ferritic/martensitic steels, activation response, mechanical property, irradiation effects

O2B.2 Low Activation Alloy V-4Ti-4Cr for Fusion and Fission Power Reactors - the RF Results

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Low activation vanadium alloys (VAs) are advanced structural materials for cores of nuclear fusion, fission and hybrid fusion-fission reactors with various coolants (Li, Na, Pb, Pb-Li, He). The main object of the R&Ds is manufacturing of VAs having functional properties (workability, welding, heat resistances of strength and corrosion) better than other structural materials. VAs will ensure realization of the complete closed nuclear cycle with structural materials reused (recycled). The referenced alloy V-4Ti-4Cr is the best alloy of V-Ti-Cr system (the USA, Japan, the RF). It is possible only small structure and composition modifications (mainly oxygen concentration) via improvements of the quality (purity, homogeneity) and thermo-mechanical treatments (TMT) of heats and articles. V-4Ti-4Cr is the high technological alloy (good rolling, tubing, welding).

The results of the RF R&Ds for the alloy V-4Ti-4Cr are presented. The alloy is manufactured and processing by JSC "VNIINM" under specification VM-DPCh-9. Ingots of weight up to 110 kg and products (tubes, plates, etc) are obtained. There are technologies for production and processing of ingots of weight 200 - 300 kg. The technologies of making and processing of the alloy and articles are aimed at optimization (minimization) of technological concentrations of impurities and optimization of the TMT regimes for a formation of bulk highly homogeneous nanostructured heterophase states ensuring enhancement of functional properties of the alloy. Nuclear physics characteristics (primary radiation damage - dpa, activation, transmutation, neutron absorption) of the alloy irradiated for a long time in neutron spectra of the RF fusion (project) DEMO-RF (15,2 dpa-Fe/year) and the fast power BN-600 (60 dpa-Fe/year) reactors are calculated and compared. The results allow to recommend alloy V-4Ti-4Cr for nuclear applications at operating temperature window 300 0C – 800 0Ñ. Continuation of the effort is needed, and must be concentrated at fusion engineering design activity and high-dose BN-600 neutron irradiation.

O2B.3 Physical properties of F82H for fusion blanket design

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Reduced activation ferritic martensitic (RAF/M) steel has been recognized as the most promising structural material for fusion blanket. Several breeding blanket concepts using the RAF/Ms have been proposed for testing in the ITER Test Blanket Module (TBM) Program. However, the design data, especially physical properties of the RAF/M are very limited. In case of F82H, developed as the 1st generation of the RAF/M steel, the published physical properties were obtained only from the F82H-IEA-heat that had previously been prepared for the round-robin testing under framework of International Energy Agency collaboration. Therefore, the material properties, focusing on the properties used for design analysis were re-assessed and newly investigated for various heats including F82H-IEA. Moreover, irradiation effects on those properties were studied in this work.

As for thermal properties, thermal conductivity that has significant impacts on the thermo-hydraulic properties of the blanket was investigated on several heats of F82H including F82H-IEA. According to the measurements, the thermal conductivity falls in the range 28.3 �} 1.1 W/m/K at 293 K. Although this is comparable with that of the other ferritic/martensitic steels, it is 20% lower than the published value for F82H-IEA. The re-assessment on the published value revealed that the thermal diffusivity was over-estimated.

As for irradiation effects on the physical properties, electric resistivity was measured after irradiation up to 6 dpa at 573 K and 673 K. The reduction of resistivity in F82H and its welds were 3% and 6%, respectively. Therefore the irradiation effects on magnitude of electromagnetic force due to plasma disruption could be quite small in the irradiation conditions studied.

16:30 – 18:00 PARALLEL M6 (Topic G: Safety Issues)
Room 5

O2C.1 Lessons learnt from ITER safety & licensing for DEMO and future nuclear fusion facilities

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One of the strong motivations for pursuing research and development in fusion energy is its potential for power production with low environmental impact and very good safety performance. For this reason, the safety and environmental (S&E) characteristics of conceptual designs of fusion power plants have been the subject of a number of studies in recent decades. It has become clear that the S&E potential of fusion can only be fully realized by careful design choices. Fundamental design decisions such as the choice of materials and of coolant can have a strong impact on the hazards that have to be taken into account. They may affect the likelihood and potential consequences of accident scenarios, environmental releases during operation, occupational radiation exposure, and the quantity and characteristics of radioactive waste at end of life.
For DEMO and other fusion facilities that will require nuclear licensing, S&E objectives and criteria should be set at an early stage and taken into account when choosing basic design options and throughout the design process.

The earlier studies of the safety of fusion power give a useful basis on which to build the S&E approach and to assess the impact of design choices. The experience of licensing ITER is of particular value, even though there are some important differences between ITER and DEMO that must not be overlooked. The ITER project has developed a safety case, produced a preliminary safety report and had it examined by the French nuclear safety authorities, leading to the licence to construct the facility. The key technical issues that arose during this process will be recalled, and the implications for the licensing of DEMO and other fusion facilities will be identified. These include safety issues that must be addressed for DEMO right from the beginning of design activities.

O2C.2 ITER tritiated waste management by the Host state

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Fusion devices like ITER will produce solid radioactive waste during operation and decommissioning. This waste results from activation by 14 MeV neutrons and contamination by tritium, used as fuel for the fusion reaction. Most of the waste will be tritiated, which requires a specific management, taking into account the physical and chemical properties of tritium, its capability to diffuse through metals and its half-life of 12,3 years (5,6 % of the tritium decay annually). France, as ITER Host state, is committed to supply "services for the management and disposal of radioactive waste arising from ITER operations".

The waste disposal facilities currently under operation in France have very strict acceptance criteria in relation to tritium content and outgassing. Therefore, and for the first time in the history of fusion research, the development of a strategy dedicated to tritiated waste management was launched in France leading to a report issued by CEA at the end of 2008 within the frame of the French national plan for the management of radioactive materials and waste. This report recommends a fifty years intermediate storage phase allowing for tritium decay before shipment to the final repository.

This paper will summarise the present status of the ITER waste management strategy under development in France, the solutions foreseen for the various waste categories and the implementation expected for the ITER tritiated waste, including considerations regarding safety. The preliminary results of an overall optimisation, based on a technical and economic approach, will be presented. The key parameters will be discussed, among which detritiation cost and efficiency, tritium waste inventory and final repository acceptance criteria.

O2C.3 Safety Issues for liquid metal blankets for ITER, FNSF and DEMO and associated R&D needs

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A leading power reactor breeding blanket design candidate for a fusion demonstration power plant (DEMO) being pursed in the US Fusion Community is the Dual Coolant Lead Lithium (DCLL) concept. The safety hazards associated with the DCLL concept as a reactor blanket during bounding accident conditions have been examined in several ARIES design studies [1, 2]. These studies both identify that the largest radiological hazards are those associated with the dust generation by plasma erosion of the blanket module first wall, oxidation of module structures at high temperature in air or steam, inventories of tritium implanted in or permeating through the ferritic steel structures of the blanket module and blanket support systems, and the Po-210 and Hg-203 produced in the PbLi breeder/coolant. What these studies lack is the scrutiny associated with a licensing review of the DCLL concept. An insight into this process was gained during the US participation in the International Thermonuclear Experimental Reactor (ITER) Test Blanket Module (TBM) Program, where, in addition to identifying the accident hazards to public and environment associated with the US DCLL TBM, issues of occupational radiation exposure, Test Blanket System (TBS) component classification and commensurate design requirements, operational and maintenance radioactive releases within the facility, and operation and decommission waste steam classifications had to be addressed. The results of this detailed safety analysis can be found in the US DCLL TBM Preliminary Safety Report (PrSR) [3]. In this paper we discuss this information in detail; and based on the lessons learned make safety proposals for the design of a Fusion Nuclear Science Facility or a DEMO that employs a lead lithium breeding blanket.

Note: This work was prepared for the U. S. Department of Energy, Office of Fusion Energy Sciences, under the DOE Idaho Field Office contract number DE-AC07-05ID14517.

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TUESDAY 17TH SEPTEMBER

08:30 – 10:30
PARALLEL T1 (Topic B: Topic Blanket Technology)
Auditorium

O3A.1 Blanket/First Wall Challenges and Required R&D on the pathway to DEMO

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The breeding blanket with integrated First Wall is the key nuclear component for power extraction, tritium fuel sustainability, and radiation shielding in fusion reactors. The ITER device will address plasma burn physics and plasma support technology, but does not have a breeding blanket. Therefore, current activities to develop "roadmaps" for realizing fusion power recognize the blanket/FW as one of the principal remaining challenges. Therefore, a central element of the current planning activities is focused on the question: what are the research and major facilities required to develop blanket/FW to the level necessary to construct and successfully operate a fusion DEMO? The principal challenges in the development of blanket/FW are: 1- The Fusion Nuclear Environment - it is a multiple-field environment (neutrons, heat/particle fluxes, magnetic field, etc) with high magnitude and steep gradients; 2- Nuclear Heating in large volume with sharp gradients – the nuclear heating drives most blanket phenomena but adequate simulation of this nuclear heating can be done only in DT-plasma based facility; and 3- Complex Configuration with Blanket/First Wall/Divertor inside the vacuum vessel – the consequence is low fault tolerance and long repair/replacement time.

These blanket/FW development challenges result in critical consequences: A- Non-fusion facilities (laboratory experiments) need to be substantial to simulate multiple fields, multiple effects, B- Results from non-fusion facilities will be limited and will not fully resolve key technical issues. A DT-plasma based facility is required to perform "multiple effects" and "integrated" experiments in the fusion nuclear environment, C- Reliability/Availability/Maintainability/Inspectability (RAMI) of fusion nuclear components is a major challenge and is one of the primary reasons why blanket/FW will pace fusion development toward a DEMO.

This paper will elucidate the top technical issues and challenges in developing blanket/first wall. We will also identify the key R&D needs in non-fusion and fusion facilities on the path to DEMO.

O3A.2 Design and Development of DEMO Blanket Concepts in Europe

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Within the EU, several blanket concepts have been considered in the past as possible candidates for DEMOnstration and fusion power plants, ranging from more conservative to higher-risk higher-payoff concepts. Among assessed breeding blanket concepts, the Helium Cooled Lithium Lead (HCLL) and the Helium Cooled Pebble Bed (HCPB) were chosen in 2002 to be tested in the European ITER TBM programme, which is expected to address many of HCLL and HCPB issues in ITER conditions.

In order to lay the foundations for a future commercial fusion power plant, a Power Plant Physics and Technology (PPPT) programme has been started in EU. The first goals are to revisit the rationale and technology development assumptions that have led to the selection of some design choices in the past, to assess their technological maturity in view of recent factual information and eventually to provide a provisional roadmap for possible realistic developments in the various areas.

Within this frame, four blanket concepts are studied including further development of the design to allow for tokamak integration as well as system level analyses. R&D and design activities are on-going in the following areas:

- R&D and design of HCPB/HCLL blanket concepts complementary to the TBM Programme.
- R&D and design of a water-cooled blanket concept with lithium lead breeding loop (WCLL).
- R&D and design of a helium-cooled concept implementing a self-cooled liquid metal breeding zone, the so-called dual coolant lithium lead(DCLL).

The issues of a WCLL and a DCLL blanket are in particular addressed in order to achieve a comparable readiness level to the HCLL and HCPB concepts and allow their consideration for DEMO.

In this paper, the present status of the four above mentioned breeding blanket concept designs will be described including their technological maturity and their development prospects. Also an overview over the main technology R&D foreseen in the PPPT blanket conceptual phase will be given.

O3A.3 Options and methods for instrumentation of Test Blanket Module Systems for experiment control and scientific mission

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The instrumentation of the HCLL and HCPB Test Blanket System (TBS) is fundamental in ensuring that ITER safety and operational requirements are satisfied as well as in enabling the scientific mission of the Test Blanket Module program. It carries out three essential functions: i) safety, intended as compliance with ITER requirements towards public and workers protection; ii) system control, intended as compliance with ITER operational requirements and investment protection; iii) scientific mission, intended as validating technology and predictive tools for blanket concepts relevant to fusion energy systems.

Instrumentation related to system control is mainly deployed in the Ancillary Systems, which are responsible to circulate fluids to and from the TBMs for cooling and tritium extraction, maintaining design nominal conditions. As such, these types of instruments are deployed from the first day of operation and foreseen for all experimental phases. Furthermore, since many are involved in sequences of operation foreseen in abnormal situations (Safety Functions) or embedded in Safety Important Components (SIC) they must satisfy requirements compliant with their safety classification, for example in term of accuracy or response time. In terms of development, control instrumentation has been identified in the pre-conceptual design activities with emphasis on adopting conventional instrumentation, which can be commercially procured and allows the assessment of reliability and maintainability required during the design review process. The main remaining research activities foreseen in the F4E TBM Program involve testing of conventional sensors (or customized versions) for specific TBS relevant conditions, which are often outside the range of certification offered by commercial vendors. The preliminary mapping of TBS control instrumentation is presented in this paper, along with the main examples of foreseen single-effect test for solicitation outside of conventional operational ranges.

Instrumentation related to the scientific mission is mainly deployed in the TBMs, although there are important exceptions such as sensors measuring tritium concentration in Pb-16Li that are installed in the liquid metal loop. These types of instruments may be deployed only for specific experimental phases as foreseen by the TBM Program test strategy, which considers to develop and install successively in ITER various TBMs specifically designed and instrumented to optimize the scope of achievable tests with respect to the different ITER operational phases. The wide range of measurements foreseen and a review of the options and methods that have been proposed as part of TBM development activities are introduced for parameters including: neutron flux; tritium generation rate; magnetic field and induced electrical currents; force/strain, displacement/position and vibration on structural components and pebble beds; temperature; pressure, flow and velocity of liquid and gaseous media; tritium concentration in, and chemical composition of, liquid and gaseous media. Further, the paper provides an overview of the research activities foreseen in the F4E TBM Program for the testing and development of the non-conventional instrumentation systems typically required as part of the scientific mission, from single-effect tests similar to those mentioned above (for example, thermal cycling for fiber-optic based sensors) to prototyping and testing in integrated fusion facilities (for example, MHD sensors). At last, the crucial challenge ofdesign integration is introduced in view of the requirements imposed by the ITER Design Review process.

O3A.4 Tritium permeation analysis in liquid metal flows with helium bubbles as applied to liquid metal breeding blankets

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In LIBRETTO test [1], evidence was obtained that helium bubbles nucleated and grew in the neutron irradiated PbLi proves. If such phenomenon occurs inside liquid metal (LM) breeding blanket channels, the study of its effect on tritium permeation and heat transfer in the near wall region will acquire utmost importance.

The T4F research group has developed in the past a nucleation, growth and transport model for helium bubbles in LM flows, as well as a tritium transport model in such a multi-fluid system [2][3]. In the present study, we are focused on the near-wall region analysis in order to obtain a wall function that allows reproducing the tritium permeation with coarse meshes and, hence, reduces the CPU time.

First, we perform some detailed CFD simulations considering different scenarios (i.e. different fraction of wall surface covered by bubbles, bubbles' diameter and geometry) and finely meshing the LM and the solid wall regions. Using this mesh, tritium diffusion processes as well as tritium recombination and dissociation are modelled. The study has been carried out using the OpenFOAM CFD toolkit. The analysis of such simulations allows us to further understand the complex phenomena and justify the use of simplified models.

Second, taking benefit of the previous simulation's results, a new model for tritium transport across a LM-solid interface partially covered by helium bubbles is developed and validated. This simplified model can be seen as a wall function for the CFD simulation which substantially reduces CPU time.

Finally, the wall function has been implemented in OpenFOAM, validated and applied to a practical case.

08:30 – 10:30
PARALLEL T2 (Topic F: Neutronics)
Room 6

O3B.1 StaTus and Verification Strategy for ITER neutronics

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It is envisaged that ITER should produce as much as 700 MW of fusion power. This equates to the production of 2.48×10^{20} 14MeV neutrons/s which will result in an uncollided flux at the first wall of approximately 4×10^{13} n/cm^2/s and a total flux of some 10^{14} n/cm^2/s with the addition of the collided component. Extensive neutronics calculations have been carried out to establish the safe operation of plant, and to determine the shielding requirements to minimise radiation exposure and show that the commitments to the Licensing Authorities in this regard are met.

The required radiation transport calculations are extremely challenging because of the large physical extent of the ITER plant, and the combination of deep penetration and streaming geometries.

At the current stage of ITER construction assumptions must be made regarding the operational scenario in order to establish the shielding requirements. These assumptions are necessarily optimistic in terms of the neutron production since we must design the shielding on the basis of the maximum level for which an operating licence is requested. Likewise, because of uncertainties in the detailed design, problems of modelling all equipment (which is so laborious as to be prohibitively time consuming), a pessimistic, or conservative, approach is adopted in shielding assessment.

Considerable expense and time is invested in designing the components to meet the requirements for safe plant operation. It is therefore important to establish at the earliest opportunity the accuracy of the neutronics calculations so that analysis can be repeated, remedial action can be taken or expense can be deferred or avoided where possible.

There are certain areas where there is a tendency to over design the plant with the accompanying detrimental effects on cost and schedule. It is therefore sensible to consider how the accuracy of the neutronics calculations causes this and define a strategy to defer expense where possible until it is established that shielding is really required.

This paper is the first attempt to define an experimental programme to establish the reliability of the neutronics calculations early in ITER operational programme. The challenge is to do this in the DD phase (or early in the DT phase) when the neutron flux will be 2 or 3 orders magnitude lower than that expected in the DT phase and the fluence will be 4 to 5 orders of magnitude lower.

The paper will summarise the current status of neutronics at ITER and then define the requirements of an experimental programme to be conducted in the early operational life-time of ITER. These will also include the use of other facilities (e.g. accelerators neutron sources, JET etc.), calibration exercises in ITER as well as measurements during the DD and DT operation.

O3B.2 Research and Development Status on Fusion DEMO Reactor Design under BA

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The goal of the DEMO reactor design under the Broader Approach (BA) is to develop possible pre-conceptual designs of DEMO by addressing key design issues and options in physics, technology and system engineering for DEMO. The joint work between EU and Japan for the DEMO design started with a benchmark of systems codes. Cross-checking between the EU systems code PROCESS and the JA systems code TPC showed a good agreement for relatively conservative plasma parameters. For extrapolation of the codes to advanced regime with higher bootstrap fraction and higher radiative fraction, appropriate models need to be developed. In parallel, critical design issues on DEMO have been studied. Difficulty in divertor heat removal is that power handling of several times as high as that in ITER with a divertor heat flux lower than ITER (10 MW/m^2 or lower). In order to resolve the problem, a reduction of divertor heat load due to plasma detachment and advanced divertor concepts such as super-X and snowflake configuration has been investigated. Regarding remote maintenance (RM), various RM concepts based on different sector segmentations and access ports has been studied to allow reasonable plant availability under severe in-vessel dose rate. The RM design study highlighted a problem that removal of the residual heat of in-vessel components in the maintenance period can be a critical issue to attain the required availability. Conceptual design of breeding blanket has been carried out in interaction with the on-going BA DEMO R&D activities on structural materials, breeders and neutron multipliers. These design studies will contribute to lining up every possible option for each component, to narrowing them down in the later phase and eventually to constructing feasible conceptual designs of DEMO.

O3B.3 Neutronic Analyses and Tools Development Efforts in the European DEMO Programme

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6. F4E, Barcelona, Spain
7. CCFE, Culham, UK
8. IIPLM, Warszaw, Poland
9. ENEA Frascati, Italy

The European Fusion Development Agreement (EFDA) recently launched a programme on Power Plant Physics and Technology (PPPT) with the aim to develop a conceptual design of a fusion demonstration reactor (DEMO) addressing key technology and physics issues.

A dedicated part of the PPPT programme is devoted to the neutronics which, among others, has to define and verify requirements and boundary conditions for the DEMO systems. The quality of the provided data depends on the capabilities and the reliability of the computational tools. Accordingly, the PPPT activities in the area of neutronics include both DEMO nuclear analyses and development efforts on neutronic tools including their verification and validation.

This paper reports on first neutronics scoping studies performed for DEMO, and on the evaluation and further development of neutronic tools. The activities were conducted by the European research associations CCFE (UK), CEA (France), CIEMAT (Spain), ENEA (Italy), HAS (Hungary), IPPLM (Poland), SFA (Slovenia), and KIT (Germany).

The nuclear analyses comprised a series of investigations to define the DEMO build and estimate the radiation loads to the TF coil, to assess the tritium breeding performance, and provide data for the nuclear heating, the radiation induced damage and helium production, as well as the activation inventory and the afterheat production. The calculations were performed with the MCNP code using a provisional DEMO model.

The tools development efforts focussed on the evaluation of available tools with the main objective to identify further development needs towards reliable, robust and flexible software tools for DEMO nuclear analyses. These included suitable Monte Carlo codes for particle transport simulations, coupling schemes of radiation transport and activation codes with the capability to provide shut-down dose radiation maps, and, geometry conversion tools to enable the import of CAD data in Monte Carlo codes.

O3B.4 An Advanced Interface Program for Multiple Neutronics and Radiation Transport Simulation Codes

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CAD/image-based Automatic Modeling Program for Neutronics and Radiation Transport (MCAM), developed by FDS team, China, is an advanced interface program between CAD systems and Monte Carlo (MC) codes. It can significantly reduce the manpower and enhance reliability for constructing MC codes input of complex systems. The last release version is MCAM 4.8. Recently, MCAM version 5.2 is going to be released. After developing various adaptors for different MC codes MCAM 5.2 is capable to convert models between multiple MC codes including MCNP, TRIPOLI, FLUKA, Geant and Super-MC. Users could analyze a complex problem with several MC codes on the exactly same model using this new function.

An advanced conversion algorithm based on face-shell shrinking was adopted in MCAM 5.2. This new algorithm can convert CAD models into better Constructive Solid Geometry (CSG) geometries with less primitive cells and less Boolean operations, which could enhance calculation performance for MC codes.

The new functions and algorithms of MCAM 5.2 were tested with the ITER Neutronics model and several other benchmark models. The results showed that the new functions and methods are efficient and available.

Furthermore, MCAM 6 is under development. A whole-body computational phantom of Chinese adult female called Rad-HUMAN was created by using MCAM from color photographic images. In the future, MCAM 6 would be able to model human body based on CT/MRI/sectioned images and nuclear facilities based on CAD models simultaneously.

08:30 – 10:30
PARALLEL T3 (Topic G: Safety Issues)
Room 5

O3C.1 Tritium safety in maintenance and waste handling in fusion reactors

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The primary objective of this presentation is to discuss tritium (T) inventory in fusion reactor materials and release during maintenance and waste handling. Materials of in-vessel components are exposed to particles of D, T, He and 14 MeV neutrons. Neutron irradiation uniformly induces radiation defects by displacements and radioactivity by transmutation. Radiation defects act as traps against hydrogen isotopes and consequently raise T inventory. If appropriate cooling is not provided during maintenance work and waste handling, temperature of materials increases by decay heat, and hence T trapped in defects starts to be released. Parameters necessary to simulate T release are temperature distribution, trap density and the activation energy for detrapping. We have evaluated trap density and the activation energy for detrapping in W and reduced activation ferritic steel F82H by neutron- and ion-irradiation experiments [1–3]. Results of simulations of T release with these parameters under several simple temperature distributions will be presented. Materials for out-vessel components are free from radiation defects and decay heat. Tritium inventory is determined by solubility of hydrogen isotopes in materials, partial pressure of T and operation temperature. The rate of T release is, in principle, controlled by surface reaction rate and diffusivity in the bulk of materials at around room temperature. In the case of austenitic stainless steel widely used as a structural material for T handling systems, it is known that the rate of T release is limited by that of bulk diffusion process under wide variety of conditions [4]. Oxide layers on surfaces, however, appear to play important roles in determination of the chemical forms of released T (HT or HTO). Recent results on T release from steels will be reviewed in the presentation.

[4] Y. Hatano et al., J. Nucl. Mater., in press, doi:10.1016/j.jnucmat.2013.01.018.
[1] Y. Hatano et al., Retention of hydrogen isotopes in neutron irradiated tungsten, Mater. Trans., in press.
[2] V. Kh. Alimov et al., J.Nucl. Mater.,in press, doi:10.1016/j.jnucmat.2013.01.208.
[3] S. Naoe et al., Fusion Sci.Technol.,**54** (2008) 515.

O3C.2 DEMO Design Activities (DDA) in the Broader Approach (BA) under Japan/EU collaboration

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Demo Design in the BA has been conducted by the DDA unit of IFERC and home teams (HTs) in EU and Japan since 2011. The activity covers most of critical issues on the DEMO design, c.f. plasma physics, divertor, in-vessel components, maintenance and safety research, etc. Work in EU-HT is conducted mainly in the EFDA Power Plant Physics and Technology (PPPT) Department and the work in Japan is conducted by Japan Atomic Energy Agency (JAEA) with many contributors from Universities, Fusion Laboratories and Industries. During the last two years, emphasis was on studies to develop the best embodiment of a tokamak as a power reactor consistent with credible operating scenarios and feasible engineering solutions to all design problems. The technical activities remain focused on the analysis of the design requirements of a DEMO reactor, on the assessment of available technical solutions and on the review of the priority R&D needs especially in the areas of the power exhaust (divertor), power extraction and tritium breeding, remote handling schemes for the internal components for high machine availability, and the development and qualification of radiation resilient structural materials. The actual work carried out, which is presented in this contribution, consisted of a number of technical activities some of which were conducted as part of a true collaborative nature such as improvement and validation of fusion reactor systems codes to define a set of machine parameters and technical characteristics of DEMO, and involved various meetings and exchange of scientific staff. Others were common studies on outstanding technical problems that are recognised to bear impact on the design of a future reactor (e.g., divertor, etc.). A collaboration of safety research has also been launched withthe goal is to develop safety design projections and eventually to compile the safety design guidelines.

O3C.3 Some technological problems of fusion materials management

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Within the framework of the International Energy Agency Programme on Environmental, Safety and Economic Aspects of Fusion Power, an international collaborative study on management of fusion radioactive materials has been carried out over the past several years to examine the back-end of the material cycle. This paper presents the results of studies performed during the last year on managing fusion radioactive materials, focusing on recycling and clearance. As clearance of sizable components (such as vacuum vessel, magnets, and bioshield) is highly desirable, we identified the source of radioisotopes that deter the clearance of these components and recommended strict impurity control and fabrication requirements. Another study was devoted to decay heat of the replaceable blanket and divertor during their periodic maintenance. A possible solution is postponement of the component maintenance for about one month after plant shut-down. This study suggests the waste management strategy in the hot cells and large waste storage facilities taking into account decay heat and tritium release problems as well as necessity of sorted storage of used tritium breeder and neutron multiplier before their reprocessing. We set forth our understanding of the main findings related to the detritiation of metals employed in plasma facing components as well as organic liquids used in experiments carried out at UKAEA, CEA-France, KIT-Germany and SCK•CEN-Belgium. Recycling (return to the production cycle) of the most scarce and expensive materials (W, V, Be, Nb, Ti) is of great interest. A special study developed the necessary techniques and processes to recycle such radioactive materials (manufacture of new components, their machining, welding, assembly, etc). The radioactivity build up in high fluence materials due to repeated use has been studied, and its effect on recycling and materials classification was evaluated. Other studies focused on the development of more complete set of recycling and clearance routes.

O3C.4 Safe Disassembly and Storage of Radioactive Components of JT-60U Torus/ Disassembly of JT-60 for JT-60SA

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Disassembly of the JT-60U torus was started in 2010 after 18-years D2 operations, and was completed in October 2012 to make the torus hall fully prepared for the installation of the JT-60SA torus. The JT-60U torus was featured by the complicated and welded structure against the strong electromagnetic force, and by the radioactiviation due to D-D reactions. Since this work is the first experience of disassembling a large radioactive fusion device in Japan, careful preparations of disassembly activities, including treatment of the activated materials and safety work, have been made. In the disassembly processes, various effective tools were utilized such as a 'diamond wire-saw without water' for cutting of hard-to-machine components to minimize dispersal of radioactive dust and eventually to minimize radioactive hazard to workers. During the disassembly period over three years, careful measures against and stringent control of radioactive exposure were implemented, and as a result, accumulated external exposure to workers of ~40000 man-day in total was 18.5 mSv, much less than the controlled target value, and no internal exposure was observed. About 12,000 elements cut into pieces with a total weight of more than 5,400 tonnes were removed from the torus hall and stored safely in a storage area. The level of radioactiviation of all disassembled components was measured and recorded with technical data such as the location in the torus hall. It was confirmed that the level of radioactiviation of the disassembly components decreases with a distance from the vacuum vessel and is almost at the background level (~0.1 micro-Sv/h) at ~10m far from the vacuum vessel for the stainless steel material. Most of the disassembly components will be treated as non-radioactive ones after the clearance verification under the Japanese regulation in future. The assembly of JT-60SA has started in January 2013 after this disassembly of JT-60U torus.

14:30 – 16:00
PARALLEL T4 (Topic B: Topic Blanket Techonolgy)
Auditorium

O4A.1 Virtual Plasma Chamber Integrated Multi-Physics Simulation: Status and Next Steps

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The geometric complexity of a fusion tokamak device makes the creation of a high performance computing simulation tool particularly attractive. This advanced, integrated simulation technique must address geometric heterogeneity and complexity, integrate multi-scales in the simulation, include multi-disciplinary models and as a result interpret phenomena from mutually dependent scientific disciplines, and predict performance using a diverse set of solvers with sufficient accuracy. The envisioned virtual plasma chamber systems package will have integrated predictive capability and be instrumental in providing information that is not easily obtainable through multi-effect physical experiments. It must be able to extrapolate performance based on a few benchmark experiments. The potential benefit of an integrated package was already highlighted in the design process for the ITER FW/blanket shield as well as test blanket module components where the goal was to reduce design uncertainty while meeting design criteria. Development of such a capability will be built upon four fundamental underpinnings. The first component in the simulation tool will be the integration and assimilation of analysis codes from the various disciplines involved. The second component involves efficient and high fidelity data mapping across various analysis codes to enable integrated or coupled simulations in a multi-physics environment. The third component is the computational analysis management, whereby all of the data relevant to each aspect of the simulation is stored and transmitted to multiple solvers in an

appropriate format, as well as made available for post-processing and debug utilities. An interactive visualization, allowing real time lifelike representation of the system response, forms the fourth component. In the past, a variety of physics codes for neutron transport, thermo-fluid MHD, mass transport, electromagnetic, and structural have been assembled and integrated utilizing manually operated interface utilities for solution exchanges. In this paper, the advantages of a virtual integrated simulation capability will be demonstrated by addressing critical issues for plasma chamber systems including tritium permeation in PbLi blankets and blanket structural behaviors under plasma transient events. Data transfer between the codes is now done manually; however, the goal is to automate this process. Undoubtedly, much greater effort will be needed to meet the full deployment of this virtual simulation capability. The steps necessary to develop an integrated analysis package regarding the aforementioned four components will be presented and discussed.

O4A.2 Facilities, testing program and modeling needs for studying liquid metal magneto hydrodynamic flows in fusion blankets

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6. ENEA C.R Brasimone, Camugnano, Italy

Liquid metals are proposed to be used as breeder materials and coolants in blanket concepts for fusion reactors since they potentially allow high thermal efficiency. Despite their attractiveness these blanket designs have still a number of feasibility issues related to the fact that the breeder or coolant is electrically conducting. When it moves under the influence of the intense magnetic field that confines the fusion plasma electromagnetic forces are induced that dominate the momentum balance in the fluid. The Lorentz forces influence significantly the flow distribution and create high pressure drop in the blanket. In order to overcome these problems or to exploit magnetohydrodynamic (MHD) phenomena, different technical solutions have been proposed whose applicability and optimization require a synergetic R&D program including dedicated experiments and complex numerical simulations.

Since many years, liquid metal MHD flows for applications to fusion blankets have been investigated worldwide. Most of the performed experiments focussed on fundamental aspects of MHD flows under very strong magnetic fields as they may occur in generic elements of fusion blankets like pipes, ducts, bends, expansions and contractions. Those experiments are required to progressively validate numerical tools with the purpose of obtaining codes capable to predict MHD flows at fusion relevant parameters in complex blanket geometries, taking into account electrical and thermal coupling between fluid and structural materials. Laboratory tests should also be performed to optimize the blanket design and to qualify measuring techniques. The aim of this paper is to highlight requirements to carry out MHD experiments and to model properly liquid metal flows in strong magnetic fields

O4A.3 Discrete element method simulations to determine the thermal impact of pebble failures as it relates to solid breeder designs

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In solid breeder blankets, major design concerns are the brittle failure of individual ceramic pebbles as well as the poor thermal transport properties of ceramic pebble beds as a whole. In previous experimental apparatuses, ceramic pebbles have been observed to 'fail' – to crack and splinter under mechanical loading. Of interest to the solid breeder design community is quantifying how accumulating failure of pebbles may impact global thermal and mechanical properties of the pebble bed. In this paper, we have approached this issue by modeling from the perspective of individual discrete spheres (referred to as 'particle dynamics' or the 'discrete element method') to allow us to gain insight into pebble failure and its consequences. With our modeling scheme we assemble a partially-periodic unit volume of solid breeder pebbles with heat transfer at two faces. We simulate the failure of a particle via a complete removal of that particle from our ensemble. Following the removal of a set number of pebbles, the remaining pebbles naturally respond (i.e. gravity and inter-particle forces induce a re-settling of the ensemble). Heat transfer through the bed is monitored during and after failure and compared to an alternate baseline case of a bed with no pebble failure. Due to the mechanics of pebble beds, thermal transport is known to proceed more readily through paths of high-stress inter-particle contacts. Depending on the location of the failed particle in relation to high-stress, three-dimensional web that runs throughout the ensemble and also how the overall structure of pebbles responds, thermal transport in the bed may be locally impaired or re-routed in measurable ways – with possible results being hot spots or an increase in thermal resistance. These two effects compound on the already-low effective thermal conductivity of the bed. In our work, we are relating the number of pebble failures (as a percent of the initial quantity) to the overall thermomechanics of the pebble beds as well as identifying a cutoff percentage above which pebble failures result in unacceptable changes in the bed. While some amount of pebble breakage is inevitable in a ceramic breeder blanket, the results of the current research will help the community develop design guidelines for robustness of pebble beds to operate satisfactorily even in the face of breakage.

14:30 – 16:00 PARALLEL T5 (Topic D: Materials)
Room 6

O4B.1 Materials R&D and materials technology for DEMO – Current level of technical readiness of candidate materials, uncertainties and strategies to close holes in knowledge

Moeslang, Anton

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O4B.2 Overview of Development on Tritium Permeation Barriers in CAEP

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It is essential to minimize the tritium release by permeation out of its confinement components or systems in the philosophy of safely handling the tritium recycling systems for a fusion reactor. The formation of a tritium permeation barrier (TPB) on the wall material is the first choice to reduce tritium permeation at high temperatures. Yet it is still under development for a practical application in the engineering. A series of fundamental investigations on oxides, aluminidmotoes, carbides and nitrides type of TPBs have been implemented in CAEP (China Academy of Engineering Physics) since 1990s. Recently, aimed at processing some real components of the tritium systems for China test blanket module (TBM) to be tested in ITER and the future tritium recycling systems in China Fusion Engineering Test Reactor (CFETR), we focused on some aluminum rich coatings of Al_2O_3/Fe_xAl_y like and some composite coatings of $Er_2O_3/Al_2O_3/Fe_xAl_y$ or $(Al,Cr)_2O_3/Fe_xAl_y$ like on a tritium containment made of stainless steel. The approaches were the hot dip aluminizing (HDA), pack cementation (PC), electrochemical plating of Al, sol-gel deposition, etc. The electro-chemically plating of aluminum from its chloride in an ionic liquid system which is composed of $AlCl_3$ and 1-ethyl-3-methylimidazolium chloride (EMIC) and later thermal oxidation to form Al_2O_3/Fe_xAl_y gradient layers was considered to be the most promising one for processing a real component with complex configuration. And it is being scaled up to some real components in China TBM tritium systems. The mechanisms of Al_2O_3/Fe_xAl_y barrier resisting tritium permeation are undergoing through the theoretical simulation and hydrogen permeation tests. And the corresponding strategies for increasing the resistance efficiency will be proposed.

O4B.3 Radiation Hardness Testing of Functional Materials for Future Fusion Reactors.

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Present research concerning ITER Diagnostics and Heating and Current Drive (H&CD) systems (to provide machine control, protection, performance evaluation and an extensive measurement capability) has raised the need to study a complete group of materials with very specific requirements: the so called “functional” materials. These components and associated materials include, among others, optical components, insulators and high frequency windows. These components will also be subjected to an intense radiation field from the ‘burning’ plasma and radiation hardness must be assessed. As many past work programmes had as main objective ITER conditions, the material testing has been reduced to quite low neutron doses. However, in the longer term beyond ITER, these materials must survive extended periods in the more hostile environment of DEMO and finally of fusion power plants (FPP). Although the doses received by functional materials will be in general lower than for structural materials, their sensitivity to radiation is also much larger (even several orders of magnitude in some cases).

As a consequence of the present draft designs for DEMO and FPP we discuss the identified R&D issues that are still required to evaluate long-term fluence or dose-related degradation of the required properties (aggregation and segregation of radiation-induced defects and impurities). Two different needs are detected. The first one is the group of present material candidates, where some radiation hardness data exist but at too low fluences. The second one is even worse, because for some materials or components, bibliographic search does not reveal relevant information even at medium doses.

To resolve these challenges, long-term research activities on functional materials must start in the short-term to evolve in parallel with DEMO designs. The types of radiation tests to be performed are reviewed. In particular, for the diagnostics and H&CD materials, in situ irradiation testing is found essential.

08:30 – 10:30
PARALLEL W1 (Topic A: First Wall)
Auditorium

O5A.1 Status of Divertor technologies and test facilities developments at IPR

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Development of technologies related to fabrication and testing of divertors for tokamak based plasma fusion devices is an important area of fusion research in India. In order to evaluate thermo-mechanical properties of materials and test the fabricated test mock-ups or components for heat transfer application, several different test/simulation facilities are also being established in India.

Divertor target test mock-ups are fabricated using high pressure sinter bonding of tungsten-alloy mono-blocks with copper-alloy tube. Integrity of the joints of these test mock-ups is checked using ultrasonic flaw detection technique.

High heat flux test facility establishment work has progressed significantly at IPR. A 200kW electron beam system capable of simulating steady-state as well as transient thermal loads on plasma facing components is procured and a large D-shaped vacuum chamber is fabricated for the test facility. The facility allows thermal load testing of components having surface area up to 1mx1m and weight up to 1 ton. Material test facilities established for thermo-mechanical testing of divertor relevant materials allow simulation of thermo-mechanical loads and measurement of thermal conduction properties of materials of interest over a wide temperature range up to 2100C. Computational software facilities are also established for relevant computational analysis and simulations.

Present paper discusses status of developments at IPR related to – (a) Ultrasonic Testing of tungsten alloy mono-block test mock-ups fabricated using high pressure sinter-bonding techniques; (b) Establishment of high heat flux test facility for thermal load testing of plasma facing components using high power electron beam system; (c) Establishment of non-destructive test facilities for testing integrity of materials and joints for heat transfer application; (d) Material test facilities for simulation/testing of thermo-mechanical properties of divertor materials over a wide temperature range; (e) Computational software facilities for simulation of coolant fluid flow and ultrasonic wave propagation.

O5A.2 Research status and issues of tungstenplasma facing materials for ITER and beyond

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Tungsten is foreseen as a plasma facing material for ITER and DEMO because of its low sputtering yield, high melting point and high thermal conductivity. Recently, status and issues of tungsten R&D as plasma facing materials for ITER have been reviewed and discussed because full tungsten divertor from the non-nuclear phase is under consideration. The important discussion points are effects of heat fluxes (steady-state and transient) and particles (especially helium ions) bombardment on surface melting and morphology changes, and their impacts on power handling capability and lifetime of the divertor.

Blistering caused by hydrogen isotope bombardment is likely not a concern because engineering surface of W mono-blocks and impurity bombardment (Be and He) would prevent their formation. He induced structures such as nano-tendrils, surface He holes, and He nanometric bubbles (formation conditions depend on mainly temperature) could be an issue only with transient heat pulses (disruption/ELM's), because the He bombardment tends to enhance pulsed heat effects shown below and transients (mainly ELM-like heat pulses) tend to induce unipolar arcing on the He induced nano-tendrils. Pulsed heat loading swith more than melting threshold energy density cause surface melting, resulting in resolidified layers with reduced mechanical integrity and reduction in power handling capability. Even for transient heat with less than the melting threshold, high cycle number of heat pulses would cause roughening, cracking and even local melting.

Among the issues remained for ITER and DEMO, effects of extreme fluence (or long discharge time) and heavy neutron irradiation are the most important. Although present ion fluence conditions (hydrogen isotopes and helium) are limited to 10^{27} m^{-2} or less, the fluence would reach 10^{29} -10^{30} m^{-2} during lifetime of ITER W divertor modules and even more for DEMO. Neutron irradiation will make the situation more complicated by changing mechanical and thermal properties of tungsten through radiation damage and elemental transmutation.

In this presentation, a summary of the review results and the discussions on present knowledge of plasma-tungsten material interactions, and additional issues for fusion reactors will be presented.

O5A.3 Modeling of MHD Issues Relevant to Liquid Lithium First Wall in China

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Liquid surface PFCs is identified as a high-risk high-reward alternative to solid PFCs in DEMO[1][2][3]. Recently, ASIPP successfully conducted an experiment of liquid lithium limiter on HT-7 Tokomak, in which a lithium film facing plasma flows under a strong magnetic field.

A lithium flow in a Tokomak is subject to the gravity, the surface tension and the Lorentz force. With temperature non-uniformly distributed along the lithium surface, Marangoni effects must be considered. We developed a solver (MHD-UCAS) based on an unstructured Cartesian adaptive system, which employed the consistent and conservative scheme to calculate the Lorentz force with good accuracy and volume of fluid method to capture the interface with good mass conservation. Special treatment is used to accurately treat the Marangoni effect. The solver has been validated and verified through comparing with analytical solutions for MHD flows in a rectangular duct and a circular pipe, with experimental data of a bubble rising in a GaInSn, with analytical and numerical solutions relevant to Marangoni effect.

The solver, MHD-UCAS, is then employed to conduct numerical simulation of lithium film flows with and without magnetic field. Solitary wave and capillary wave are well illustrated. MHD effects and Marangoni effects on lithium film flows are analyzed. Lithium bubble splashing on a lithium film is also studied, in which the bubble has an initial velocity. The multiscale phenomena of MHD flows will be shown. It will be concluded to form a quasi-steady lithium film flow in Tokomak facing plasma, we must consider all of the effects from the gravity, the surface tension, the Lorentz force and also the Marangoni effect and the effect of plasma currents.

O5A.4 Developments with explosive forming for complex sheetmetal components and with explosive bonding as a joining technique for dissimilar materials for ITER

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The ITER project requires development of new manufacturing techniques.

Explosive Forming was developed for forming large, complex geometries such as the vacuum vessel of ITER and the magnet coil cans for MAST.

Also new applications with Explosive Bonding were developed for joining many dissimilar materials in different geometries.

CuCrZr was explosive bonded to 316L stainless steel for realizing tube transitions. The CuCrZr was applied in aged condition which was maintained during bonding. Tensile and helium leak tests showed superior properties of the interface. CuCrZr plates were bonded to stainless steel in solutionized condition. Aging is realized during a cequential HIP cycle. An interlayer of pure copper or pure nickel prevented a significant increase of chromium at the interface due to diffusion that would occur during this cycle.

Copper was explosive bonded around obstacles on the stainless steel Triangular Support for the vacuum vessel. The copper was clad from several plates that overlap at the edges; at the overlaps a sound Cu-Cu bonding was realized.

Molybdenum-to-copperbonded plates were machined and formed after explosive bonding.

Heat treatments and rolling processes were developed for optimizing surface quality and for reducing internal elastic stresses. Thermal fatigue tests were successful. A local heat load with power density in the order of 100 MW/m2 caused melting and cracking of the molybdenum surface but did not damage the molybdenum-copper interface.

For WEST PFCs, Tungsten was explosive bonded to CuCrZr with a copper interlayer which enabled the interface to survive a water quench after heating to 500 °C.

Other new applications are under development such as explosive bonding copper to beryllium.

This contribution shows that new applications with the explosion technologies were developed for meeting the challenges of fusion reactors. More applications are to be expected.

This work was partly set up in collaboration and financial support of F4E.

08:30 – 10:30
PARALLEL W2 (Topic F: Neutronics)
Room 6

O5B.1 Fusion Yield measurements on JET and their Calibration

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The power output of fusion experiments and fusion reactor-like devices is measured in terms of the neutron yields which relate directly to the fusion yield. In this paper we describe the devices and methods used to make the new in-situ calibration of JET in March 2013 and its early results.

The target accuracy of this calibration was 10%, just as in the earlier JET calibration and as required for ITER, where a precise neutron yield measurement is important, eg for tritium accountancy.

We discuss the constraints and early decisions which defined the main calibration approach, e.g. the choice of source type and the deployment method.

We describe the physics, source issues, safety and engineering aspects required to calibrate directly the Fission Chambers and the Activation System which carry the JET calibration. In particular a direct calibration of

the Activation system was planned for the first time in JET. We used the existing JET remote-handling system to deploy the 252Cf source and developed the compatible tooling and systems necessary to ensure safe and efficient deployment in these cases.

The scientific programme has sought to better understand the limitations of the calibration, to optimise the measurements and other provisions, to provide corrections for perturbing factors (e.g. presence of the remote-handling boom and other non-standard torus conditions) and to ensure personnel safety and safe working conditions. Much of this work has been based on an extensive programme of Monte-Carlo calculations which e.g. revealed a potential contribution to the neutron yield via a direct line of sight through the ports which presents individually depending on the details of the port geometry.

This work, part-funded by the European Communities under the contract of Association between EURATOM/CCFE was carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission. This work was also part-funded by the RCUK Energy Programme under grant EP/I501045.

*See the Appendix of F. Romanelli et al., Proceedings of the 24th IAEA Fusion Energy Conference 2012, San Diego, USA

O5B.2 Development of Super Monte Carlo Simulation Program for Fusion and Fission Applications

Song, Jing[1,2]; Sun, Guangyao[1,2]; Zheng, Huaqing[1]; Chen, Zhenping[1,2]; Wu, Mingchang[1,2]; Hao, Lijuan[1]; Li, Gui[1]; Chen, Cheng[1,2]; Zhang, Kuan[1,2]; He, Tao[1,2]; Yu, Shengpeng[1,2]; Zeng, Qin[1]; Long, Pengcheng[1]; Hu, Liqin[1,2]; Wu, Yican[1,2] and FDS Team[1,2]

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2. University of Science and Technology of China, Hefei, Anhui, China

The Super Monte Carlo Simulation Program (SuperMC) for fusion, fission and other nuclear applications has been developed by FDS Team in China. Radiation transport, isotope burn-up, material activation, radiation dose and biology damage simulation can be performed by MC and coupled deterministic methods. Many recent advances in the MC methodology and advanced computer technology are incorporated in SuperMC, developing advanced method exclusively for nuclear applications. Complicated geometries and the whole physical process of various types of particles in broad energy scale can be well handled. The techniques of variance reduction and hybrid parallel computing were implemented in SuperMC to enhance efficiency. SuperMC is implemented in an object-oriented programing language C++ with modular design concept, and is relatively easy to maintain, modify and expand. Bi-directional automatic conversion between general CAD models and full-formed input files of SuperMC is supported by MCAM. Mixed visualization of dynamical 3D data set and geometry model is supported by RVIS. Continuous-energy cross section data from nuclear data library HENDL are utilized to support simulation. Comprehensive design and analysis for fission and fusion reactors can be performed on SuperMC by cloud computing service.

SuperMC 2.2, the latest version, can perform neutronics fixed source and critical design parameters calculations for reactors of complex geometry and material distribution based on the transport of neutron and photon. It can be used for fuel burnup analysis based on a built-in depletion equation solver. The constructed solid geometry and optimized algorithms based on solid hierarchy were mainly employed. Complete physical processes including photonuclear reaction and up-scatter effect of thermal neutron were accounted for. The simulation efficiency was enhanced by rich variance reduction techniques and hybrid parallel on particles and space decomposition based on MPICH and OpenMP. The systematic and intelligent simulation from automatic modeling to online calculation visualization can be performed with the support of MCAM, RVIS and HENDL.

The correctness of SuperMC2.2 has been verified through calculations conducted on the ITER reference neutronics model. Nuclear heat and fast neutron fluence rate in TF inboard leg under four different conditions and nuclear heat of blanket modules were calculated and the results were in accordance with MCNP results. SuperMC based reactor physics design and analysis have been also performed for China LEad-Alloy cooled Reactor (CLEAR), which is designed and constructed by FDS Team.

O5B.3 Automatic Mesh Adaptivity for Hybrid Monte Carlo/Deterministic Neutronics Modeling of Fusion Energy Systems

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2. University of Wisconsin-Madison, Madison, Wisconsin, United States of America

Over the last decade, the role of neutronics modeling of nuclear energy systems has been shifting from separate analyses of individual components to high-fidelity, full-scale analyses of the entire systems. The high accuracy associated with minimizing modeling approximations by including more physical and geometric details is now feasible because of advancements in computing hardware and development of efficient modeling methods. The CADIS and FW-CADIS hybrid Monte Carlo/deterministic techniques dramatically increase the efficiency of neutronics modeling, but their use in the accurate neutronics analysis of fusion energy systems has been limited by the large computational requirements for their preliminary deterministic calculations and final Monte Carlo calculation. Three mesh adaptivity algorithms were developed to reduce the memory requirements of CADIS and FW-CADIS without sacrificing their efficiency improvement. First, a macromaterial approach enhances the fidelity of the deterministic models without changing the mesh. Second, a deterministic mesh refinement algorithm generates meshes that capture as much geometric detail as possible without exceeding a specified maximum number of mesh elements. Finally, a weight window coarsening algorithm decouples the weight window mesh and energy bins from the mesh and energy group structure of the deterministic calculations in order to remove the memory constraint of the weight window map from the deterministic mesh resolution. The three algorithms were used to enhance an FW-CADIS calculation of the prompt dose rate throughout the ITER experimental facility. Using these algorithms resulted in a 23.3% increase in the number of mesh tally elements in which the dose rates were calculated in a 10-day Monte Carlo calculation and, additionally, increased the efficiency of the Monte Carlo simulation by a factor of at least 3.4. The three algorithms enabled this difficult calculation to be accurately solved using an FW-CADIS simulation on a regular computer cluster, obviating the need for a world-class super computer.

O5B.4 TRIPOLI-4 version 9 Shielding for the fusion community: overview, relevant benchmarking for fusion and licencing policy

Trama, Jean-Christophe; Brun, Emeric; Damian, Frederic; Diop, Cheikh; Dumonteil, Eric; Hugot, Francois-Xavier; Jouanne, Cedric; Lee, Yi-Kang; Malvagi, Fausto; Mazzolo, Alain; Petit, Odile; Visonneau, Thierry; Zoia, Andrea

CEA Saclay, Gif-sur-Yvettes, France

TRIPOLI® represents a family of radiation transport codes using the Monte Carlo method. It has been continuously developed at CEA since the mid 60s. TRIPOLI-4®, being the fourth generation, has been developed starting from the mid 90s in C++. It tracks neutrons from 20 MeV down to 10-5eV, photons, electrons and positrons from 100 MeV to 1 keV, in 3D arbitrary geometries either in TRIPOLI® proprietary format or ROOT© format. It reads its continuous energy nuclear data from any evaluation in ENDF format, including but not limited to JEFF-3.1.1 and ENDF/B-VII.0. It has various simulation modes, fixed-source (with or without fission), criticality, or both of them chained. A variety of tallies are available: averaged flux in a volume (track length and collision estimators), surface flux and current, point flux, reaction rates, deposited energy, dpa, pka, gamma spectroscopy, mesh tallies, tallies for criticality (keff, beta eff, kinetic parameters). It comes with embedded response functions from IRDF and EAF as well. Several variance reduction techniques, including automated ones, make it particularly suitable to deep penetration problems (implicit capture, Russian roulette, exponential biasing with automatic estimation of the importance map). It may run in parallel, on single multi-core machines as well as heterogeneous networks of stations or massively parallel machines. It comes with productivity tools designed to speed-up the elaboration and verification of the input deck as well as the output analysis: 2D interactive display of geometry; 2D display of mesh tallies and collision sites; 3D input deck modeler through SALOME with CAD import capability; compatibility with ROOT® tools to post-process TRIPOLI® outputs.

Its V&V test base is made of more than 1000 internationally accepted benchmarks from ICSBEP and SINBAD and covers all its application domains, including several fusion cases (D-T neutron time-of-flight, ITER Neutron Wall Load, FNG, ITER Bulk Shield Experiment etc.) which will be detailed in the ISFNT paper.

TRIPOLI-4® is the reference Monte Carlo code for CEA, EDF, and a number of branches of AREVA. A “Shielding-only” version has been designed to specifically answer the needs of the fusion community. It will be distributed through the NEA Databank with a licence for R&D, covering the fusion applications, whose terms will be detailed in the paper.

08:30 – 10:30
PARALLEL W3 (Topic A: First Wall)
Room 5

O5C.1 Fusion Technology Aspects of Laser Inertial Fusion Energy (LIFE)

Meier, W.R.; Dunne, A.M.; Kramer, K.J.; Reyes, S. and the LIFE Team

Lawrence Livermore National Laboratory, P.O. Box 808, Livermore, CA 95550, USA

Experiments are being conducted at the National Ignition Facility seeking to demonstrate fusion ignition with net energy gain. If these experiments are successful, it would create the potential for commercial laser inertial fusion energy (LIFE). Conceptual design and supporting R&D activities are being led by Lawrence Livermore National Laboratory and involve a highly diverse and multidisciplinary set of participants, including representatives from the electric utility industry, power plant equipment vendors, the National Laboratories, academia, the business and financial community and state and federal regulators. This talk provides an overview of the LIFE power plant design but focuses on the fusion engine requirements and designs.

As with all IFE chambers, the LIFE chamber must deal with the short-range emissions from the target, which includes x-rays and energetic ions. The LIFE approach is to use a low density Xe fill gas to absorb the ions and most of the x-rays thus spreading out the delivery time (and thus peak heat load) to the first wall. The LIFE chamber is modular and is located within a separate vacuum chamber. This configuration was chosen to allow rapid maintenance since the radiation damage life of near-term reduced activation ferritic/martensitic (RAFM) steels cannot be accurately predicted at this time. Liquid lithium is used as the first wall and blanket coolant. Lithium's attractive features include low mass density (leading to lower chamber mechanical loads), good heat transfer characteristics, good tritium breeder capability, and affinity to retain tritium. The high tritium solubility greatly reduces tritium permeation and helps limit the site inventory. A molten salt tritium extraction technique, previously demonstrated at Argonne National Laboratory, is the baseline tritium recovery approach. The estimated plant tritium inventory is about 500 g, with the highest contribution from the target manufacturing system. To avoid the possibility of Li contact and reaction with water, the LIFE design includes an intermediate heat transfer loop using a molten salt similar to the heat transfer salts used in the solar thermal industry.

The power conversion cycle is based on the steam Rankine cycle that has a well-established industrial based. The thermal conversion efficiency for LIFE with a blanket outlet temperature of 575 ºC is estimated to be about 44%. Over the next decade, efficiency improvements (to 50% or more) may be possible by going to supercritical steam cycles that are currently being developed worldwide. This, however, will require development of higher temperature fusion materials such as advanced ODS steels.

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

O5C.2 Physics and Technology for Engineering and Power Plant in Laser Fusion Energy Systems under Repetitive Operation

Perlado, J. Manuel[1]

1. Instituto de Fusión Nuclear (DENIM)/ETSII/Universidad Politécnica, Jose Gutierrez Abascal, 2, 28006 Madrid, Spain
and teams in HiPER Project and Virtual Reactor Modelling
2. Departamento Ingeniería Energética, ETSII, UNED, Madrid
3. Academy of Science-PALS, Czech Republic
4. Comissariat à l'Energie Atomique et aux Energies Alternatives, France
5. STFC Rutherford Appleton Laboratory, Didcot, UK

Different proposals of Laser Fusion Energy have been envisioned in the last years. Those concepts cover from Engineering Facilities at large scale in Energy, to Power Prototyping and final DEMO Reactors. HiPER (Europe), LIFE (USA) and LIF_T (Japan), and other Chinese or Russian initiatives, are entering a new phase where is critical the integration of systems (lasers, target design, manufacturing and injection, chamber and blanket, tritium handling and Power cycles). The study of an Engineering Burst (in HiPER) facility with a repetitive laser operation, with realisticburst mode of hundreds to thousands of shots at 5-10 Hz rate, and small gain under continuous (24/7) repetition (Prototype) or final high gain Demo Reactor is finished and presented indicating its importance. It could be important the difference between Prototype and Demo, because the different target energy gains could have consequences in the first wall and optics. The Engineering Test Bed results could be able to demonstrate with the lowest risk**,** repetitive laser-injection systems in an already defined model of Chamber. Assuming those conditions, we review work on how could be possible to accommodate Experiments in Technology relevant for Prototype and Reactor. This paper will show the differences in designing the facility for single shot operation (NIF or LMJ Ignition/Gain machines), or repetitive systems, from Laser requirements to Chamber area, activation and damage in optics, wall and structural materials and also dose assessment. A summary of the differences in new designs from Engineering, Prototyping and Demonstration approaches will be the key goal. It will presents integration of the various systems needed for an early demonstration of laser-driven power production, including the requirements of tritium and neutron breeding, use of existing reactor-capable materials and considerations of plant safety. The approach here presented uses two levels: research in each one of the key questions from fundamental to applied physics and technology; and integration of our available answers into a Power Plant System for defining progressive designs from burnup: to thermo-mechanical responses of materials; fluidynamics; tritium generation and cycle; accident analysis after evaluation of activation and radionuclide concentrations, safety and radioprotection. That is a virtual reactor modelling integrating systems with help of specific proposed experiments.

O5C.3 Closing the fusion fuel cycle: LIFE tritium recovery and processing

Reyes, Susana[1]; Anklam, Tom[1]; Becnel, James[2]; Dunne, Mike[1]; Farmer, Joe[1]; Bandhauer, Todd[1]; Kramer, Kevin[1]; Martinez-Frias, Joel[1]; Miles, Robin[1]; Taylor, Craig[3]; Shanahan, Kirk[2]

1. Lawrence Livermore National Laboratory, Livermore, California USA
2. Savannah River National Laboratory, Aiken, SC, USA
3. Los Alamos National Laboratory, Los Alamos, NM, USA

Laser Inertial Fusion Energy (LIFE) is designed to deliver a transformative source of safe, secure, sustainable electricity, in a time scale that is consistent with the global energy market needs. The LIFE market entry plant is being designed to demonstrate the feasibility of a closed fusion fuel cycle, including tritium breeding, extraction, processing, re-fueling, accountability and safety, in a continuously operating power-producing device. While many fusion plant designs require large quantities of tritium for start up and operations, a range of design choices made for the LIFE fuel cycle act to substantially reduce the in-process tritium inventory. The high fractional burn-up in a LIFE capsule greatly relaxes the tritium breeding requirements, while the use of only milligram quantities of fuel per shot and choice of a pure lithium heat transfer fluid substantially reduce the amount of tritium entrained in the facility. Additionally, the high solubility of tritium in the lithium is calculated to mitigate the need for development of permeation barriers in the engine systems, normally required to protect against routine releases. A methodology for recovery of the tritium fuel from the blanket via a solvent extraction process is being investigated, with various potential technology solutions under evaluation.

The present paper offers an overview of the design of the LIFE fuel cycle, including the technology challenges and the development pathway proposed.

(Work performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract DE-AC52-07NA27344.)

O5C.4 Development of fusion-fission hybrid reactor utilizing high power/ rep-rate lasers

Lee, BongJu[1]; Kong, Hong Jin[2]; Lee, Jae Young[1]; Park, Young Chun[1]; Kim, Young Gil[1]

1.Handong Global University, Pohang, S. Korea
2. KAIST, Daejeon, S. Korea

We have developed laser fusion energy technology and fundamental technology for sustainable fusion-fission hybrid reactor. The features of stable atomic fusion energy, SAFE are:
-Utilizing high power/rep-rate laser, which is self-navigation laser fusion driver using SBS-PCM method.
-Utilizing technologies for target production and injection, which establishes production technology for less than 1 cent per target and develops tracking technology for targets injected at 400m/s 15Hz.
-Utilizing low cost laser diode, which is inexpensive high-efficient and stacked array.
-Utilizing fission generator module surrounding the fusion chamber, which may provide the ultimate slolution for blanket. Above four technologies play an important role of achieving the commercially available fusion energy.

THURSDAY 19TH SEPTEMBER

PLENARY SESSION 2
08:30 – 10:30
Auditorium

I2.1 China's Plan for Design and R&D Activities of Multi-Functional Fusion Test Reactor

Wu,Yican, FDS Team

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Fusion energy is one of the most important ways for finally solving mankind's demand for energy without planet-warming gases, radioactive-waste headache. China is keen to promote fusion energy research and has launched series design and R&D activities yet.
Proposed roadmap and testing strategy of fusion energy will be presented based on the assessment and design analysis of the developed series fusion reactors and hybrid system concepts in China as well as a summary of related design and R&D activities in the world. In this development strategy, based on wide survey and analysis of development of the international fusion reactor and related research by FDS Team, a China Fusion DEMO Reactor (C-DEMO) concept has been proposed. To check and validate the C-DEMO relevant technologies based on the viable technologies, a fusion engineering experimental reactor will be built in China. Recently a series of scenario options for test reactor concept have been proposed, such as the Multi-Functional hybrid eXperimental reactor (MFX) concept and the China Fusion Engineering Test Reactor (CFETR) concept. In addition, this strategy also considered the possibility of accelerator driven system (ADS) as a key nuclear technology test facility to promote fusion development.

In this contribution, a summary of development strategy of fusion energy, currently proposed design concepts, required key technologies and the overall R&D activities progress in China are presented.

I2.2 Development of MW-range fusion neutron sources on the roadmap to DEMO

Kuteev, B.V.

National Research Centre "Kurchatov Institute", Moscow, Russian Federation

Reaching the demonstration level of fusion reactions for the energy production and non-electrical applications remains the major goal of magnetic fusion development. The DEMO development line is aimed at a GW fusion power plant based on fusion physics and technologies tested within ITER project. The necessity of research supporting the ITER-DEMO line in the field of fusion nuclear science and technology is evident. Therefore several facilities with the fusion power in the range of 1-1000MW are currently considered by the magnetic fusion community. Those aimed to solve physical-technical problems of DEMO and to demonstrate device operations in the domain interesting for energy or non-energy applications.

The MW-range fusion neutron sources using neutral beam driven fusion may be built on the basis of tokamak, stellarator, RFP or mirror machine concepts. Such devices are attractive due to their lower cost compared with ITER and may reach the level of neutron loadings of 0.2-3 MW/m^2.

The expected input of the MW-range FNS line in the DEMO development is as follows:

-reduction of the commercialization time due to offering neutrons as the final product of controlled fusion, which may restore the credibility of controlled fusion;

-creation of the competition environment for fusion nuclear science & technology development in low cost devises;

-initiation of the full set of steady state fusion technologies compatible with neutron environment, which are needed for DEMO;

-providing appropriate fusion neutron spectra for fusion materials and components development;

-tests of DEMO-ITER tritium breeding modules in a specialized facility.

The MW-range FNS are useful in basic research being supplementary to fission reactors and spallation neutron sources for neutron scattering and may stimulate the development of fusion-fission hybrid systems and technologies.

I2.3 A Fusion Nuclear Science Facility for a Fast-Track Path to DEMO*

Garofalo, A.M[1]; Abdou, M[2]; Canik, J.M[3]; Chan, V.S[1]; Morley, N.B.[2]; Sawan, M.E.[4]; Taylor, T.S.[1]; Wong, C.P.C.[1] and Ying, A.[2]

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3. Oak Ridge National Laboratory, Oak Ridge Tennessee 37830, USA
4. University of Wisconsin-Madison, Madison, Wisconsin 53706, USA

An accelerated fusion energy development program, a "fast-track" approach, requires proceeding with a nuclear and materials testing program in parallel with research on burning plasmas, ITER. Proceeding with a Nuclear Fusion Science Facility (FNSF) in parallel with ITER provides a strong basis to begin construction of DEMO upon the achievement of $Q \sim 10$ in ITER. A FNSF would address many of the key issues that need to be addressed prior to DEMO including breeding tritium and completing the fuel cycle, qualifying nuclear materials for high fluence, developing suitable materials for the plasma-boundary interface, and demonstrating power extraction. The Advanced Tokamak (AT) is a strong candidate for an FNSF as a consequence of its mature physics base, capability to address the key issues, and the direct relevance to an attractive target power plant.

Key features of AT are fully noninductive current drive, strong plasma cross section shaping, internal profiles consistent with high bootstrap fraction, and operation at high beta, typically above the free boundary limit, $\beta_N > 3$. Recent research shows that full noninductive and high β_N scenarios are obtained and sustained for many energy confinement times. Key remaining challenges are to sustain these AT scenarios for several current redistribution times, τ_R, and develop high fluence boundary solutions consistent with high plasma performance.

A moderate sized FNSF-AT has an advantage of limited tritium consumption, robust tritium self-sufficiency (achievable TBR > ~1.1), and sufficient neutron flux (2 MW/m^2) to test components. An example design point gives a Cu-coil device with $R/a =$ 2.7 m/0.77 m, k = 2.3, $B_T = 5.4$ T, $I_p = 6.6$ MA, $\beta_N = 3.7$, $P_{FUS} = 230$ MW, and $P_{COILS} = 400$ MW. The finite aspect ratio provides space for a solenoid and robust plasma current initiation. The modest bootstrap fraction of $f_{BS} \sim 0.75$ provides an opportunity to develop steady state with sufficient current drive for adequate control.

*This work was supported by General Atomics IR&D funding and the US Department of Energy under DE-FG02-08ER54984, DE-AC05-00OR22725, and DE-FG02-09ER54513.

11:00 – 13:00
PARALLEL TH1 (Topic A: First Wall)
Auditorium

O6A.1 Design, Fabrication and Testing of the ITER FW and Shielding Blanket

Raffray, R.[1]; Calcagno, B.[1]; Chappuis, P.[1]; Dellopoulos, G.[2]; Eaton, R.[1]; Fu, Zhang[1]; Gervash, A.[6]; Chen, Jiming[3]; Kim, D. H.[4]; Kim S. W.[4]; Khomiakov, S.[5]; Labusov, A.[6]; Martin, A.[1]; Merola, M.[1]; Mitteau, R.[1]; Ulrickson, M.[7]; Zacchia,F. and all contributors to the BIPT* effort

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2. F4E, EU ITER Domestic Agency, Barcelona, Spain
3. SWIP, China ITER Domestic Agency
4. NFRI, ITER Korea
5. Open Joint-Stock Company "N.A. Dollezhal Research and Development Institute of Power Engineering" (NIKIET), 107140, Malaya Krasnoselskaya Str. 2/8, Moscow, Russian Federation
6. Federal State Unitary Interprise Efremov Scientific Research Institute of Electrophysical Apparatus (NIIEFA Efremov), 3, Road to Metallostroy, Metallostroy, St Petersburg, 196641, Russian Federation
7. SNL (US ITER Domestic Agency), Albuquerque, NM, USA

The main functions of the blanket system are to:contribute in absorbing radiation and particle heat fluxes from the plasma; contribute in providing thermal shielding to the vacuum vessel, diagnostics and external vessel components; and provide a plasma- facing surface compatible with the plasma performance requirements and a limiting surface defining the plasma boundary during limiter operation and plasma start-up/ramp-down. The blanket system comprises Blanket Modules (BM) segmented into 18 poloidal locations. Each BM consists of two major components: a plasma-facing First Wall (FW) panel and a Shield Block (SB). The BM is attached to the vacuum vessel through a mechanical attachment system of flexible supports and keys.

The blanket design process is very challenging due to demanding design and interface requirements and constraints. These include high heat fluxes from the plasma, large electromagnetic loads during off-normal events and the need to provide sufficient contribution to the shielding of the vacuum vessel and superconducting coils.

A well-planned R&D program provides a key complementary element to the design process to help validate the design performance in the midst of all the interface constraints. A major part of this R&D is the prequalification for Procurement of the Blanket SB and FW. The FW prequalification process includes small scale mock-up manufacturing and heat flux test leading up to a semi-prototype prequalification stage, whose successful completion would confirm acceptance of each procuring Domestic Agency as a FW supplier for ITER. A pre-qualification program was also implemented for the SB providers, requiring that each procuring Domestic Agency manufactures a Full Scale Prototype (FSP) based on one of the SBs that would be part of their final supply.

This paper describes key aspects of the blanket design, highlights solutions developed to accommodate demanding interface requirements, and summarizes the pre-qualification programs of both the FW and SB.

**The ITER Blanket design and analysis effort is being conducted through the Blanket Integrated Product Team (BIPT) which consists of members from the ITER Organization and from Domestic Agencies (China, European Union, Korea, Russian Federation, USA)*

O6A.2 R&D and procurement status on ITER divertor components in JAEA

Suzuki , Satoshi; Ezato, Koichiro; Seki, Yohji ; Yokoyama, Kenji; Mohri, Kensuke; Kakudate, Satoshi

Japan Atomic Energy Agency, Naka-Shi, Ibaraki-Ken, Japan

JAEA is developing divertor components for fusion devices, mainly for ITER and JT-60SA. JAEA, assigned as a Japanese domestic agency (JADA) to implement ITER activity, has concluded a procurement agreement with ITER Organization (IO) to make in-kind procurement of the divertor outer vertical target (OVT) components for ITER. The 4-staged procurement activity of the OVT components is currently in the first stage. We have been manufacturing an OVT full-scale prototype since 2009 with a lot of help and support by IO. The first high heat flux test of the plasma-facing unit (PFU#1) of the full-scale prototype, which was manufactured by mid of 2011, has been carried out in ITER Divertor Test Facility (IDTF) in Efremov Institute in 2012 under the procurement agreement between IO and Russian domestic agency (RFDA). This test campaign has successfully completed under close collaboration with IO/RFDA/JADA. In particular, tungsten-armored part of the PFU#1 showed excellent thermal performance and durability in this test. It has survived cyclic heat flux of 10 MW/m2 for 5,000 cycles and 20 MW/m2 for 1,000 cycles without degradation of its cooling capability, which demonstrates that JADA's current manufacturing technology has sufficient technical level to manufacture fully tungsten-armored divertor target for ITER.

In parallel to this procurement activity mentioned above, JAEA is extensively developing fully tungsten-armored divertor target under an R&D task agreement which was entered into force at the end of 2012. JAEA has already provided 12 small-scale divertor mock-ups with tungsten monoblock armor. These mock-ups will be high heat flux tested in JAEA's facility and in IDTF to qualify the joining technologies to be applied in the manufacturing of a full-scale fully tungsten-armored prototype.

In this paper, JAEA's R&D and procurement activities mainly on the ITER divertor components are summarized.

O6A.3 Qualification of ITER EHF First Wall in Russia

Gervash, A.[1]; Eaton, R.[2]; Glazunov, D.[1]; Kuznetsov, V.[1]; Mazul, I.[1]; Mirgorodsky, V.[1]; Mitteau, R.[2]; Raffray, R.[2]; Rulev, R.[1]; Semichev, D.[1]

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2. ITER Organization, Route de Vinon sur Verdon, 13115 Saint Paul Lez Durance, France

The ITER First Wall (FW) design has been considerably changed in the last years because of the necessity to improve resistance to electromagnetic loads and to adapt FW performance to severe (up to 5 MW/m2) surface heat loads. As a consequence, it has been necessary to upgrade or re-develop a number of technologies associated with the previous design. In particular, explosion bonding and Hot Isostatic Pressure (HIP) methods were experimentally checked to provide a vacuum tight CuCrZr/SS bimetallic boundary for the hypervapotron (HV) type water cooling channel. To attach individual FW elements (fingers) to the supporting structure (beam) with the required strength but minimum shrinkage, a unique welding technology (weld thickness 15 mm) based on power fibre laser was developed and experimentally checked. This paper describes the above mentioned technologies and gives examples of their experimental approbation.

Along with a design and analyses of the FW panels, some aspects of performance can only be validated by a “design by experiment” approach. This paper describes the “design by experiment” campaigns undertaken to determine some key parameters for armour tile size and HV cooling channel heat sink performance.

To validate the thermo hydraulic design of the hypervapotron water cooling channel with respect to the occurrence of the critical heat flux (CHF) event, (burnout – loss of water containment) a number of representative mock-ups were manufactured and tested at High Heat Flux (HHF). This paper presents the main results of the test campaign that demonstrates the required cooling efficiency and critical heat flux margin (1.4) at a water velocity of ≥2 m/s.

To establish an acceptable beryllium armour tile size for the Enhanced Heat Flux (EHF) FW panels, an experimental campaign was performed. This paper gives the results of the HHF experiments where the tile size was varied in the range 12x12x6 - 50x50x8 mm. It is shown that optimization of the tile geometry and joining technology provides the required cyclic fatigue lifetime for the reference EHF FW design.

In conclusion, this paper describes a large scale FW semi-prototype, sequence of its fabrication and further HHF testing programme.

O6A.4 The WEST project: testing ITER divertor high heat flux component technology in a steady state tokamak environment

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ITER baseline plans the use of a full tungsten (W) divertor for the nuclear phase and discussions to start ITER operation with the full W divertor are ongoing. However, the required high heat flux technology has never been tested in the demanding environment of a tokamak under the steady state plasma heat fluxes expected in ITER (10-20 MWm^{-2}). In order to mitigate the risks for ITER, it has been decided to equip Tore Supra with a full W divertor, benefitting from the unique long pulse capabilities of the Tore Supra platform, the high installed power and the long history of experience with actively cooled high heat flux components at the IRFM.

The transformation from the current circular limiter geometry of Tore Supra to the required X-point configuration is achieved by installing a set of copper poloidal coils inside the lower and upper parts of the vacuum vessel. The new configuration will allow for H-mode access, providing relevant plasma conditions for plasma-facing component technology validation.

The lower divertor target design is closely based on ITER design (W monoblocks, same cross section), while the upper divertor region usea heat sink technology similar to the one adopted for the ITER blanket modules (CuCrZr copper/stainless steel) with a tungsten coating (in place of the Be tiles which ITER will use).

Four years are expected to be required for the manufacture of the W divertor elements. This industrial-scale manufacturing (~500 plasma facing unit, ~14% of ITER divertor surface) and the associated quality assurance testing will alone be of great use to ITER. The power handling capability of the first units could be tested in tokamak from late 2015 on while the complete divertor should be available in 2018.

The final divertor configuration will complement existing W divertor experiments in Europe (e.g. JET and ASDEX-Upgrade) by adding the important component of long pulse capability and actively cooled surfaces to the operational experience being gathered elsewhere. Extended plasma exposure provides access to ITER-critical issues such as PFC lifetime (melting, cracking etc.), tokamak operation on damaged metallic surfaces, real time heat flux control through PFC monitoring, fuel retention and dust production etc.

11:00 – 13:00
PARALLEL TH2 (Topic H: FNT – Special Neutron Sources)
Room 6

O6B.1 IFMIF, a fusion relevant neutron source for material irradiation: current status

Knaster, J.; Chel, S.; Fischer, U.; Groeschel, F.; Heidinger, R.; Ibarra, A.; Micciche, G.; Möslang, A.; Sugimoto, M.; Wakai, E.

A fusion reactor aims at generating energy by fusing deuterium and tritium nuclei to He^4; the by-productis a 14.1 MeV neutron. The nascent He nuclei (3.5 MeV) are trapped by the electromagnetic fields in the plasma chamber contributing to the plasma heating; in turn, the produced neutrons collide and react with the nuclei of the first wall, the blanket, the radiation shield and the vacuum vessel, and thereby are used for breeding tritium and generate heat for electricity production. Such nuclear interactions cause structural damage deteriorating the mechanical properties and the functionality of the structural materials. In ITER, the maximum expected displacements per atom (dpa) at the end of its operational life of the steel-based first wall will be less than 2 dpa. However, a fusion power plant can accumulate up to 150 dpa within 5 years full power operation. The understanding of the material degradation exposed to such extreme damage levelsis essential for the engineering design and licensing of any fusion power facility.

The related nuclear processes include the generation of Primary Knock-on Atoms (PKA) affecting the lattice structureas well as nuclear reactions leading to transmutations. Due to the high PKA energy, compared with the average displacement energy threshold for the production of Frenkel vacancy-interstitial defect pairs typically in the order of some 10 eV, a cascade of recoiling atomsis initiated through elastic collisions. In turn, transmutation reactions such as (n,α) with a reaction thresholdabove a few MeV neutron energy, lead to swelling and embrittlement through the accumulation of the He atoms in the lattice defects. In fission neutron sources, with an average neutron energy below 2 MeV, the fusion relevant He production rates (10 appm He/dpa) are missed by about 2 orders of magnitude and are therefore not suited to study theHe induced embrittlement. In turn, spallation neutron sources, with high energy tails in the range of100 MeV, can produce 100 appmHe/dpa and also generate by transmutation significant amounts of light elements that affect significantly the mechanical properties of the structural materials. Accordingly mission-tailored neutron facilities that adequately simulate fusion relevant conditions are indispensible for establishing the required data basis.

The d-Li based International Fusion Materials Irradiation Facility (IFMIF) will provide sufficiently high neutron intensity with a suitable neutron spectrum to fulfil the materials testing requirements. The IFMIF project is presently in its Engineering Validation and Engineering Design Activities (EVEDA) phase under the Broader Approach (BA) Agreement between Japan Government and EURATOM. EVEDA aims at the construction and testing of the most challenging facility sub-systems, such as the a first accelerator stage, the lithium target and loop, and irradiation test modules, as well as the design of the entire facility, thus to be ready for the IFMIF construction with a clear understanding of schedule and cost at the termination of the BA mid-2017. The paper will review the IFMIF facility and its principles, report on the status of the EVEDA activities and achievements.

O6B.2 Technical analysis of an early fusion neutron source based on the enhancement of the IFMIF/EVEDA accelerator prototype

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2. CIEMAT, Madrid, Spain

In the framework of the Engineering Design and Engineering Validation Activities for the International Fusion Materials Irradiation Facility (IFMIF/EVEDA), 3 major prototypes have been designed and are being manufactured, commissioned and operated:

- the Accelerator Prototype (LIPAc) at Rokkasho, fully representative of the IFMIF low energy(9 MeV) accelerator (125 mA of D+ beam in continuous wave), now under construction andplanned to be commissioned by June 2017;
- the Experimental Lithium Test Loop (ELTL) at Oarai, integrating all elements essential to the IFMIF lithium target facility, already commissioned in February 2011;
- critical components of the two High flux test modules currently tested in the HELOKA-LP helium loop at Karlsruhe.

In this paper, possibilities are analysed, from a technical point of view, for combining, modifying, and enhancing at limited cost, selected components of the prototypes towards the realisation of an early reduced-power neutron source, able nonetheless to start the testing of candidate DEMO materials, while the decision for the construction of the complete IFMIF plant may be under consideration.

Various options of deuteron beam parameters, such as energy, current and shape are analysed with respect to their technical challenges and the neutron yield resulting from the nuclear reaction with the Li target. Related requirements for the liquid Li target with respect to jet size, velocity and curvature are evaluated and the neutron mapping in the high flux area is presented underlying an analysis of the available volume for testing ferritic martensitic steels at relevant damage rates, such as 5, 10 and 15 dpa/year.

O6B.3 FAFNIR: strategy and risk reduction in accelerator driven neutron sources for fusion data

Surrey, E.; Porton, M.and FAFNIR collaboration members

EURATOM/CCFE, Culham Science Centre, Abingdon, UK

The 14MeV neutron source has two purposes:

(i) populate the materials database with engineering relevant information,

(ii) provide 14MeV irradiation data to validate and calibrate more readily available fission and ion irradiation data so to strengthen predictive modelling capability.

Demanding the qualification of materials to first wall lifetime equivalent 150dpa results in the IFMIF specification, a technically challenging, high risk and expensive facility. In consequence no timetable is foreseen that will deliver materials testing data commensurate with the start of EU DEMO construction in 2030.

Applying “defence in depth” strategies to the design circumvents the 150dpa requirement for in-vessel materials, a previously perceived necessity for regulatory licensing. Instead, the function of material testing becomes investment protection and engineering design. This logic provides a prioritized list of functional requirements:

(i) identify new damage phenomena associated with 14MeV neutrons

(ii) calibration and validation of data from fission and ion irradiations

(iii) validation of materials towards lifetime damage levels

Many degradation phenomena manifest above 10dpa, so the required neutron intensity is relaxed, enabling less challenging options to be considered. One such concept is proposed: FAcility for Fusion Neutron Irradiation Research. FAFNIR is a 40MeV, 5-30mA cw D^+ beam incident on a rotating multi-layered carbon target. The on-target beam area maximizes irradiation volume for relevant dpa rates, giving 25cm^3 at 4-20dpa and 150cm^3 at 1-5dpa, whilst multi-scale sample analysis is exploited to maximize the population of the volume. The accelerator is less challenging than IFMIF and the target, although requiring some R&D for the higher current, is an extrapolation of existing systems. Issues related to liquid target cavitation and the need for precise beam profiling are eliminated. Such a facility could deliver useful data within 8 years of project start and initiate populating the database of fusion relevant irradiation to inform DEMO design development.

O6B.4 Feasibility Study of an Intense D-T Fusion Source. "The New Sorgentina"

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The international community agrees on the importance to build a large facility devoted to test and validate materials to be used in harsh neutron environments. The IFMIF project is still under discussion and, in any case, its realization will take several years.

A facility whose dimension and cost are much reduced with respect to IFMIF is proposed by ENEA. The facility reconsiders a previous study known as "Sorgentina" but takes into account new technological development so far attained.

The "New Sorgentina" Fusion Source (NSFS) project is based upon an intense D-T 14 MeV neutron source with T and D ion beams impinging on a 1 m radius rotating target. NSFS produces an high neutron flux (> 5x1013 n/cm2/s) over about 250 cm3. Larger volumes of lower neutron flux will be available (e.g. for TBM experiments) as well as collimated channels to study some features of the ITER neutron camera. A major improvement is the enhanced neutron production attainable by using a pair of faced rotating targets with the high intensity flux region in between.

One of the key issues is the ions source and the accelerating system. The NSFS facility will overcame this problem by means of an upgraded version of the well known JET-PINI as ion beams.

This facility can be realized in a few years provided upon launching in advance a preliminary technological programme to address some technical problems e.g. the study of ion source in continuous mode, the target heat loading and removal, target and tritium handling and inventory as well as site licensing.

The proposed facility has to be realized by ENEA and should be considered an European Facility. Beside the reduced cost, data attainable from this facility could represent a valid reason for justifying its construction. The main characteristics of NSFS project will be presented.

11:00 – 13:00
PARALLEL TH3 (Topic I: Repair & Maintenance)
Room 5

O6C.1 The European contribution to ITER Remote Maintenance

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2. ITER Organization, Route de Vinon sur Verdon, 13115 Saint Paul Lez Durance, France

For a first-of-a-kind nuclear fusion reactor like ITER, remote maintainability of neutron-activated components is one of the key aspects of plant design and operations, and a fundamental ingredient for its success and for the demonstration of long-term viability of fusion as energy source.

The European Domestic Agency (EU DA) has always contributed to building up a credible remote maintenance scenario and to defining an effective remote handling set-up for ITER, by means of design and R&D activities, and now that the ITER construction has begun its role is even greater.

In fact, important support is being given to the ITER Organisation in specifying the functional requirements of the so-called Remote Handling Procurement Packages (i.e. the subsystems allocated to EU DA belonging to the overall ITER Remote Maintenance Systems IRMS), and in performing design and R&D activities – with the support of national laboratories - in order to define a sound concept for these packages.

Furthermore, domestic industries are being involved in the subsequent detailed design, validation, manufacturing and installation activities, in order to actually fulfil our procurement-in-kind obligations.

After an introduction to ITER remote maintenance, this paper will present the current status and the next stages for the remote handling systems to be procured by EU DA (that represents a significant fraction of the whole IRMS):

- Divertor remote handling;
- Cask and plug remote handling;
- In Vessel Viewing System;
- Neutral Beam Cell remote handling

and will also illustrate complementary aspects related to cross cutting technologies like radiation tolerant components and remote control systems.

Finally, the way all these efforts are coordinated will be presented together with the overall implementation scenario and milestones.

O6C.2 Verifying and validating ITER divertor Remote Handling on Divertor Test Platform

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The lower part of the ITER reactor is divertor. The divertor operates under the heaviest particle loading in the reactor. Eventual replacement of the divertor is necessary due to the erosion of divertor components facing plasma, and/or to allow development of the divertor design. The remote replacement of the entire divertor system is estimated for three times in the first 20 years of ITER operation.

Replacement of the ITER divertor is foreseen to take place a few times during the ITER lifetime, and is therefore classified as an "RH Class 1" activity. All such operations must be designed in detail, and shall be verified prior to ITER construction. In the case of divertor maintenance, the feasibility of planned principles and equipment is done on a full scale physical test facility, DTP2 (Divertor Test Platform), which is located in VTT facility, Tampere, Finland. Work is carried in seamless cooperation by VTT and TUT researchers.

The DTP2 consists of a full-scale mock-up of the lower section of the ITER divertor (30 deg. of the thorus), a divertor element transporter (Cassette mover), various tools used for the cassette connecting and disconnecting, a water hydraulic manipulator arm for handling the tools, the mover control hardware and the operator control room. The DTP2 (Divertor Test Platform) is a full scale test facility, where the in-vessel divertor maintenance principles, equipment and control methods are tested and developed further. Besides testing the divertor maintenance cycle and developing the related operational tasks, the DTP2 facility is used to verify/improve the design of the divertor components, and of the related interfaces.

The divertor maintenance includes the cassette transportation and positioning, locking/unlocking the cassette into the reactor, the cassette cooling circuit disconnecting/connecting and other operations like dust cleaning. All these operations from the reactor port opening to closing it again will be taken care by the divertor RH -maintenance system.

This paper discusses the DTP2 activities in RH system processes, mechanical, control and security aspects. Presented are results obtained this far and planned DTP2 extensions and future test plans. Also the biggest challenges foreseen in ITER divertor Remote maintenance are discussed.

O6C.3 R&D status on Remote Maintenance Technology in JAEA

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Procurement activity for the blanket remote handling system (BRHS) concluded with a procurement agreement (PA) with ITER organization (IO) in Dec. 2011. The Japanese domestic agency (JADA) which was assigned to implement ITER activities by JAEA has already begun developing the BRHS. These activities required functional specifications that met with the system requirements document (SRD) and will progress to the detailed design phase in the beginning of 2014 and fabrication & integration tests in the beginning of 2020, respectively.

JADA has carried out R&D to confirm compatible design feasibility for remote rescue in the case of a BRHS malfunction, and also for remote decontamination of activated tungsten dust in order to meet the major requirements of the SRD.

Failure criticalities against failure events for remote rescue were classified, and the effects of such events on the BRHS were assessed by a failure mode and effects analysis (FMEA). From this analysis, a rescue scenario using a redundant screw mechanism driven by an external system was designed based on critical failure modes, such as an electric failure due to insulation deterioration, mechanical failure due to adhesion, or seizure due to gamma radiation. The design feasibility of the remote rescue was confirmed by an interference check for verifying accessibility using a VR demonstration and distinct element method for the redundant mechanism.

For remote decontamination, a grease lubrication which catches dust easily is not used for the exposed parts such as gears, linear guide, and ball screw mechanisms. It is also difficult to apply to the rack & pinion and linear guide mechanisms. Therefore, dry lubricants such as diamond-like carbon (DLC) or molybdenum disulfide films were designed and tested under real loads. From the results of the endurance tests, the fact that the dry lubricant films will require a high wear and abrasion resistance was confirmed.

In this paper, JAEA's R&D to verify the design feasibility of the ITER BRHS is summarized.

O6C.4 Pre-conceptual Design Assessment of DEMO Remote Maintenance

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EDFA, as part of the Power Plant Physics and Technology (PPP&T) programme, has been working on the specification and pre-conceptual design of a Demonstration Power Plant (DEMO). It is intended that DEMO will take fusion from experimental research to demonstrating the feasibility of commercial power generation. As part of this programme, in 2011 a review of the remote maintenance strategy was conducted. The review considered maintenance solutions compatible with expected environmental conditions (i.e. radiation and decay heating), and that at the same time showed potential for meeting the plant availability targets. A key finding of the resulting assessments was that for practical purposes the expected radiation levels prohibit the use of complex remote handling operations to replace the power plant first wall.

In 2012 the activities in the area of DEMO remote maintenance were further extended, providing an excellent insight into the requirements, constraints and challenges of fusion power plant maintenance. In particular, the assessment of blanket and divertor maintenance, in light of the expected radiation conditions and availability, has elaborated the need for a very different approach from that of ITER. Furthermore, it is seen that constraints in terms of vacuum vessel port size and shape will now be an important factor in driving the overall machine design (i.e. a critical input into System Codes regarding machine geometry and tritium breeding blanket design). This activity has produced some very informative virtual reality simulations of the blanket segments and pipe removal that are exceptionally valuable in communicating the complexity and scale of the required operations. Through these detailed simulations, estimates of the maintenance task durations has been possible. The results suggest that for a highly developed and tested maintenance system, with a large element of parallel working and challenging but feasible operation times, the full replacement of the blanket and divertor components could be achieved within a time frame of 6 months.

In support of the conceptual maintenance operations, a first indication of the size and the requirements of an Active Maintenance Facility (AMF) have been elaborated. The AMF will be used for storage, testing, handling and processing of in-vessel components, when these are not installed in-vessel. In addition, it will need to provide for the maintenance and testing of the multiple complex remote handling and break-down recovery systems needed to ensure timely extraction of the in-vessel and near-vessel components.

This paper will provide an overview of the pre-conceptual design assessment of the power plant maintenance paying particular attention to the in-vessel operations and active maintenance facility.

Key words: DEMO, PPP&T, Remote Handling, Maintenance, Divertor, Blanket.
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14:30 – 16:00
PARALLEL TH4 (Topic A: First Wall)
Auditorium

O7A.1 Manufacturing and joining technologies for helium cooled divertors

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In the helium cooled (HC) divertor, developed at KIT for a fusion power plant, tungsten has been selected as armour as well as structural material due to its crucial properties: high melting point, very low sputtering yield, good thermal conductivity, high temperature strength, low thermal expansion and low activation. Thereby the armour tungsten is attached to the structural tungsten by thermally conductive joint. Due to the brittleness of tungsten at low temperatures its use as structural material is limited to the high temperature part of the component and a structural joint to the reduced activation ferritic martensitic steel EUROFER is foreseen. Hence, to realize the selected hybrid material concept reliable tungsten-steel and tungsten-tungsten joints have been developed and will be reported in this paper.

In addition the modular design of the HC divertor requires tungsten armour tiles and tungsten structural thimbles to be manufactured in high numbers with very high quality. Due to the high strength and low temperature brittleness of tungsten special manufacturing techniques need to be developed for the production of parts with no cavities inside and/or surface flaws. The main achievement in developing the respective manufacturing technologies will be presented and discussed.

To achieve the objectives mentioned above various manufacturing and joining technologies are pursued. Their later applicability depends on the level of development including their transferability to the component. Hence, specifying design and requirements for the components of interest will determine appropriate time and criteria for selecting most promising technologies. Although the considered technologies are mainly developed for the HCdivertor it is worth to note that they are also useful for other divertor and even blanket concepts, particularly those with tungsten armour.

O7A.2 Current status of W/Cu divertor for EAST and related R & D for actively cooled HHF components

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With rapid increase in H & CD power on EAST, much higher heat fluxes goes to the divertor. The bolting connection between tiles and heat sink cannot meet the requirements to remove the heat fluxes away. Therefore a W/Cu divertor project has been launched at ASIPP, aiming at achieving actively-cooled full W plasma-facing components (PFCs) for the upper divertor around 2013-2014 with a heat removal capability up to 10 MW/m^2, followed by the lower divertor around 2014-2015. And the passive plates, part of the first wall (FW), will also be changed into actively-cooled W/Cu-PFCs around 2013-2014, together with the upper divertor.

The PFCs for vertical targets have ITER-like monoblock configuration, being manufactured by hot isostatic pressing (HIP) for cladding oxygen free Cu (OFC) to the inner surface of the W monoblocks, and then HIP or hot radial pressing (HRP) for the bonding between the clad monoblocks and CuCrZr cooling tube. The PFCs for DOME and FW will be flat-type, being manufactured by three different processes. The first is OFC casting onto the rear side of W tiles firstly, followed by vacuum hot pressing (VHP) of the W/OFC tiles onto CuCrZr heat sink plate. The second is HIPing process, also making use of cast OFC as the interlayer between W and CuCrZr. The third is CVD-W coating onto an OFC layer diffusion-bonded to CuCrZr plate by means of VHP.

An ultrasonic non-destructive testing (NDT) method has been developed to examine the bonding quality of OFC clad W monoblocks, and the method is now being extended into the curved CuCrZr tube of the monoblock PFCs. Unique NDT methods are under development for the coming batch production, including phased array ultrasonic (PAU) and eddy current testing (ECT), to inspect the products more efficiently and costly.

O7A.3 Status and prospects of the EU development of the He-cooled divertor for DEMO power plant

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The helium-cooled divertor development has been pursued at the Karlsruhe Institute of Technology (KIT) within the frame of the earlier European Fusion Power Plant Conceptual Study with the goal of reaching a performance of 10 MW/m2. In the course of conceptual development, the He-cooled modular divertor with jet cooling (HEMJ) design has been defined as the reference concept [1] in the early 2000s. It is based on a modular design of small tungsten-based cooling fingers. Each of them consists of a tile as thermal shield which is brazed to a thimble as heat sink part; both are made of tungsten material. The fingers are connected with the steel body by means of brazing and are cooled with helium impinging jets at 10 MPa and 600°C. For a proof of design principle, a combined high heat flux (HHF) test facility (electron beam with helium loop) was built in Efremov, St. Petersburg, Russia in 2004. It allows for mockup tests without neutrons under the specified DEMO conditions. Already in the first HHF test series, the cooling performance of the finger concept with helium under the heat load of 10 MW/m2 has been confirmed. Since then, the design was improved in terms of robustness (i.e. thermal stress reduction) and the production quality of tungsten parts to enhance the lifetime of the cooling fingers against thermo-cyclic loads. The first breakthrough was achieved in 2010 when such a cooling finger of the currently best design already survived > 1000 cycles at 10 MW/m2 without damage. Current R&D focuses on the manufacturing technology of multi-finger modules and their integration onto the target plate with additional study of the edge finger modules. The former includes, among others, high-temperature brazing and mass production (e.g. by deep drawing) of tungsten divertor components. To ensure quality, non-destructive testing (NDT) of the cooling finger modules was studied at the SATIR facility, CEA, using the same method as for the ITER divertor tests. First NDT tests have already shown promising results. This paper reports on progress to date of the overall development.

1] P. Norajitra, et al., Progress of He-cooled Divertor Development for DEMO, Fusion Eng. Des. 86 (2011) 1656–1659.

14:30 – 16:00
PARALLEL TH5 (Topic D: Materials)
Room 6

O7B.1 Materials R&D for a timely DEMO: key Findings and Recommendations of the EU Roadmap Materials Assessment Group.

Stork, Derek and the EU Roadmap Materials Assessment Group [1]

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Alongside the recent EU Fusion Roadmap exercise [2] the EU Fusion Programme established an independent 'Materials Assessment Group' (MAG) to: assess the state of R&D on structural, Plasma Facing (PF) and High Heat Flux (HHF) materials for DEMO; identify the major knowledge gaps and risks; recommend a coherent strategy and road map, bearing in mind the needs to mitigate development risks; and define a resource-loaded R&D plan up to a decision point to construct DEMO in the early 2030s.

The Group's findings and recommendations, as incorporated into the Roadmap, are discussed. A systems-engineering approach to the analysis of materials assemblies is emphasised considering: whole-system reactor operational cycles; needs for efficient maintenance and inspection; and interaction with functional materials/coolants. A key requirement is the assessment of materials by their Technology Readiness Levels (TRLs). There are important lessons to be learned from Fission reactor material development, especially in safety and licensing, fabrication/joining techniques and development of manufacturing and supply-chain.

The Group has identified 'baseline' material(s) for each DEMO role, and established the risks to the achievement of the DEMO mission from the known limitations, or unknown properties, associated with each baseline material. R&D programmes to address these risks are then proposed. The DEMO mission assessed is the Roadmap's choice of a DEMO phase I with a 'starter blanket' and 'starter divertor': the blanket being capable of withstanding at least 2MW.yr.m^{-2} fusion neutron flux (equivalent to ~20 dpa damage in its front-wall steel). The Roadmap then foresees a second phase for DEMO with in-vessel front-wall steel loading up to at least 50 dpa. The MAG recommendations for baseline materials for DEMO Phase I are the 'EUROFER' Reduced Activation Ferritic-Martensitic (RAFM) steel for blanket structure, tungsten for PF and Divertor armour and copper alloys for the water-cooled HHF Divertor sub-structure. Each of these materials still has significant associated risks however, so, MAG recommends a parallel development of 'Risk Mitigation Materials'(RMM): ODS RAFM steels and High temperature Ferritic-Martensitic steels for the blanket structure and a range of advanced tungsten laminates, fibre reinforced tungsten or copper and functionally-graded tungsten-copper for the Divertor. The total R&D programme is composed of parallel tests and development of the baseline and RMM, with appropriate 'down-selection' points to align with the need to fix the engineering definition of the DEMO Blankets and Divertors. This programme is integrated with the ITER Test Blanket Module programme developments.

The use of ITER licensing experience to refine the issues in nuclear testing of materials is addressed. This results in limits on the scope of materials tests with fusion neutron ('14MeV') spectrum before DEMO design finalisation, but, nevertheless some testing with 14MeV neutrons is essential to Fusion Materials development. To do this in a timely manner requires deployment of a ≥ 30 dpa (steels) 14MeV testing capability by 2026. The optimisation of the testing programme by the pre-testing with fission neutrons on isotopically - or chemically-doped steels and with ion-beams is discussed along with the minimum requirements of the 14MeV testing programme itself. The technical possibilities for early deployment, based on variations and acceleration of the IFMIF programme are assessed.

Fundamental, and mission-oriented modelling of the nuclear reactions and damage to materials from intense neutron fluxes is essential for the whole development programme and the recommendations for major streams of this theoretical effort are also discussed.

[1] Members of the Assessment Group: D Stork, P Agostini, J-L Boutard, D Buckthorpe, E Diegele, S L Dudarev, C English, S Gonzalez, A Ibarra, C Linsmeier, G Marbach, B Raj, M Rieth, M-Q Tran and S J Zinkle

[2] F. Romanelli, The European Fusion Roadmap, *this conference.*

O7B.2 Current Status of the Technology Development for Fabrication of Indian Test Blanket Module (TBM) of ITER

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1. Indira Gandhi Centre for Atomic Research, Kalpakkam, India
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India has successfully developed Indian RAFM (INRAFM) steel for fabrication of Indian Test Blanket Module and at present extensive characterization of this material is in progress. For fabrication of TBM, the technologies being considered are Hot Isostatic Pressing (HIP) to produce first wall of TBM, and technologies of Gas Tungsten Arc (GTA), Electron Beam (EB), Laser and Laser Hybrid welding. Welding consumables for joining this steel has been developed and characterized. Properties of the GTA welds met the entire specification requirements comparable with that of the base metal. This consumable has also been successfully used to carry out Hybrid laser welding of RAFM steel.

The procedure for EB welding to join plates of thicknesses up to 12 mm has been developed. Impact tests conducted on EB welds showed that toughness of the weld metal in the as-welded condition is comparable to that of the base metal. A box structure, which simulates one of the components of TBM, has been fabricated using EB to demonstrate the applicability of the process for component fabrication.

Laser welding of 6 mm thick plates of RAFM steel has also been carried out successfully and the properties of the weld joints have been found to be satisfactory. Procedure for laser hybrid welding of 12 mm thick RAFM steel is being developed.

Initial trials carried out has demonstrated that plates with internal channels, as required for the first wall of TBM can be produced by joining plates with pre machined grooves by Hot Isostatic Pressing. Channel collapsing during HIPing is overcome by inserting leachable ceramic core. Trials are in progress also with alternate techniques to develop procedures to produce first wall of TBM using HIP process.

This paper will present the current status of the development of fabrication technologies and future plans to realize the fabrication of Indian TBM.

O7B.3 The Progresses in SiC/SiC Component Development Fusion/Fission Application

Kohyama,Akira and Kishimoto, H.

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2. IEST Co., Ltd., Tokyo, Japan
3. GUNZE Co., Ltd., Kyoto, Japan

Based on the steady progress in Ceramic and Carbon materials R & D, the role of those high temperature non-metallic materials is becoming more important for the advanced nuclear reactor systems. Where, fiber reinforced ceramic composite materials have been recognized to be more attractive options for nuclear fission and nuclear fusion energy systems. .

Among the attractive potential candidates, SiC/SiC composite materials have been recognized as one of the most attractive options, especially for blanket and divertor of fusion reactor and reactor core components of LWR and other fission reactors.

However, the activities on SiC/SiC for fusion and even for fission have been limited within quite small scale efforts, comparing with those efforts on ferritic/martensitic or stainless steels.

The main reason has been the scale difference in overall industrial activities between SiC/SiC and high temperature steels.

This paper provides the recent progresses in SiC/SiC development toward early utilization for LWRs and Fusion DEMO.

After the March 11 Disaster in East-Japan ensuring safe technology for LWR becomes a top priority R & D in nuclear energy policy of Japan. Along this line, replacement of Zircaloy claddings to SiC/SiC based fuel cladding is becoming one of the most attractive options and MEXT fund based projects, such as SCARLET*1 by the authors group and another by Toshiba, and METI fund based projects, such as INSPIRE*2 by the authors group and another by Toshiba, have been launched as 4 to 5 year termed projects. These projects care for NITE process and CVD/CVI process for making long SiC/SiC fuel pins and connecting technology integration. The outline and the present status will be briefly introduced.

SiC/SiC for fusion application is slowly but steadily expanding including first-wall structural component application, flow channel insert application and W-SiC/SiC divertor applications.

As the part of Broader Approach project between Japan and EU, SiC/SiC R & D for IFERC program and SiC/SiC heater development for IFMIF/EVEDA program are on-going.

Those activities are strongly related with large scale production capability of SiC fibers and SiC nano-powders and SiC/SiC fabrication process line establishment including CVD/CVI process facility upgrading. Those will be also introduced.

*1: SiC Fuel Cladding/Assembly Research Launching Extra-Safe Technology

*2: Innovative SiC fuel Pin Research

14:30 – 16:00

PARALLEL TH6 (Topic I: Repair & Maintenance)
Room 5

O7C.1 Automated in situ line of sight calibration of ASDEX Upgrade bolometers

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The IPP ITER Bolometer Group recently developed the ITER Bolometer Robot Test Rig (IBOROB). It is a robot-based diagnostic tool, which allows the measurement of the lines of sight (LOS) of the ITER bolometer prototypes. Until now it was only used for the collimator development [1].

IBOROB was further developed and can now be operated in ASDEX Upgrade during the vessel opening. At present, once a diagnostic like the bolometry is mounted inside the vessel, the actual LOS are not measured in situ, but only derived from CAD. This procedure allows now the fully automatic three dimensional measurement of the bolometer LOS.

Due to the high position accuracy of the robot, the system has a high spatial resolution. The absolute accuracy, in reference to the tokamak coordinate system, is provided by an additional calibration procedure with a measurement arm by FARO Technologies.

The main goals are to measure the spatial distribution of the bolometer channels and to identify the amount of misalignment. Specific camera type dependencies and occlusions of single LOS by e.g. blanket modules can also be qualified.

The method and results obtained in a first set-up in ASDEX Upgrade are presented. In addition they will be compared with ex situ measurements in the laboratory to approve the results. The paper describes the different steps needed to apply the measurements in the vessel and focuses on the constraints, e.g. geometrical ones, for an application in a tokamak.

This technique gives new perspectives on maintenance and operating activities for an experiment like ITER. Future requirements, for making the robot a routine tool for tokamak maintenance, will also be discussed. The paper concludes with a summary of key recommendations to improve the diagnostic tool.

References:

[1] H. Meister, F. Penzel et al.: Development of an automated method for insitu measurement of the geometrical properties of the ITER bolometer diagnostic, Fusion Engineering and Design 86 (2011) 1170-1173.

O7C.2 Mobile robot for the inspecting in vacuum vessel of ITER

Wu, Huapeng

Lappeenranta University of Technology, Finland

The inspection of the vacuum vessel (VV) of the ITER from inside or outside is very difficult because of various constraints, such as non-magnet effect material, high temperature, constrained space, radiation etc. Therefore, special robots are required for the inspecting tasks. In the assembly of the VV, the NTD test carries out after welding, and a film need to be brought to the other side of the welded wall to capture the X-ray and detect the defect-welded points. As space is very limited, the current solution is to build up a tracker on the back of the wall, in order to guide a robot to the welding gap. However, it costs very much and affects the function of the VV, due to the extra tracker on the back of the wall. This paper presents a special mobile robot, called as the spider-robot, which can freely run on the non-magnet material surface in any position within a confined space. The robot is designed based on the principle of vortex, and its configuration and control are presented in the paper and the potential applications for the ITER are addressed. The robot is able to carry out the NTD test for detecting the welding and inspecting the first-wall surface.

O7C.3 In-Vessel Components Water Cooling Pipes: Design, Installation and Maintenance Strategy

Martin, Alex; Dell Orco, Giovanni; Escourbiac, Frederic; Furmanek, Andreas; Gicquel, Stefan; Jokinen, Tommi; Macklin, Brian; Merola, Mario; Palmer, Jim; Raffray, Rene

ITER, St. Paul lez Durance, France

The ITER Tokamak Cooling Water System (TCWS) provides coolant for blankets and divertor. The blanket system consists of 440 blanket modules (BMs). The blanket manifold consists of a system of seamless pipes arranged in 20 degreee bundles and routed from the upper ports of the Vacuum Vessel (VV) to the blanket modules. In each upper port there is an inlet and an outlet bundle of 20 pipes, which split at the port exit in 2 branches supplying either the inboards or the outboard blankets. The manifold is routed between the VV and BMs. Branch pipes provide the connection between the cooling circuits in the modules and those in the manifold through a coaxial connector welded to the shield block. A complex, sequential installation sequence has been developed in order to enable the assembly. Supporting R&D has been launched to validate this process. Once installed the manifold is considered a semi-permanent component, but since failure would prevent ITER operation a maintenance strategy has been planned. The divertor system consists of 54 independent cassettes. Each of the divertor cassettes is fed via two independent coolant feed and return pipes, welded to matching pipe stubs protruding from the divertor cassette. 54 pipes are routed through the top section of the nine lower ports (6 pipes per port). The remaining 54 pipes are routed through dedicated vessel penetrations. The installation of the divertor pipes is relatively straight forward, but the divertor cassetes require scheduled maintenance. A strategy for cutting and rejoining divertor pipes and a detailed concept of the equipment required for fully remote maintenance has been developed. It is planned to launch supporting R&D to validate this concept.This paper describes the design for assembly and maintenance of the blanket manifold and the divertor pipes and the status of the supporting R&D.

FRIDAY 20 TH SEPTEMBER

08:30 – 10:30
PARALLEL F1 (Topic H: FNT)
Auditorium

O8A.1 Wendelstein 7-X: Status of Project Construction and Commissioning Planning

Gasparotto, M.; Baylard, C.; Boscary, J.; Bosch, H. S.; Brakel, R.; Hartmann, D.; Grote, H.; Klinger, T.; Lorenz, A.; Nagel, M.; Naujoks, D.; Peacock, A.; Rummel, T.; Schauer, F.; Stadler, R.; Vilbrandt, R.; Wegener, L.

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The stellarator device Wendelstein 7-X (W7-X) is now under the final stage of assembly. All
the machine components have been built and in mid 2014 the commissioning activities should start.

The first objective of W7-X is to prove the stellarator optimisation principles, i.e., to reach the same confinement quality as a similar sized tokamak. The second objective is to demonstrate stable high-power steady-state operation.

W7-X is a superconducting fusion device with a five-fold magnetic field periodicity which implies five nearly identical modules, each consisting of two flip-symmetric half modules with five different non-planar and two different planar coils each.

The main parameters of W7-X are: major radius 5.5 m; minor radius 0.53 m; magnetic fieldon plasma axis 2.5 – 3 T; plasma volume 30 m^3; pulse length 1800 sec (at 10 MW of heating power); total mass 750 t; cold mass 425 t.

In the present paper, after a short description of the device, the ongoing activities to complete
the construction will be summarised, focussing on the optimisation of the layout of the torus hall and on the completion and assembly of the in-vessel components.

The main lessons learned during the assembly will be presented as well as the application of the quality control, the handling of non-conformities and the design change procedures. The planning of the commissioning sequences to perform the integral tests of the major W7-X systems is discussed in order to prepare the W7-X device to start plasma operation. The following elements are analysed: (i) the objectives of the tests, (ii) the adopted methodology, and (iii) the risks of failure and repair actions.

O8A.2 Helical Reactor Design FFHR-d1 and c1 for Steady State DEMO

Sagara, A.; Miyazawa, J.; Goto, T.; Tamura, H.; Tanaka, T.; Yanagi, Y. and FFHR Design Group,

National Institute for Fusion Science, Toki, Gifu 509-5292, Japan

Towards DEMO, the Fusion Engineering Research Project (FERP) has been launched in NIFS since 2010, by starting the re-design of the LHD-type helical reactor FFHR, namely a demo version "FFHR-d1". In the first round, main parameters have been selected (major radius of the helical coils R_c = 15.6 m, toroidal field B_c = 4.7 T, plasma volume V_p ~ 2,000 m^3, fusion power P_f= 3 GW), using the design window analyses under the key constraints of the space between the superconducting (SC) magnet and the plasma (minimum 0.89 m at the inboard side), the magnetic stored energy (maximum 160 GJ) and the neutron wall loading lower than 2 MW/m^2. The second round is ongoing for detailed 3D designs of the SC magnet support-structures, the large maintenance ports, the in-vessel components of blanket and divertor, and 3D neutronics analyses. The divertor targets in FFHR-d1 are located behind the blankets so as to avoid the direct neutron irradiation. This mitigates the constraints on material selection for divertor targets. In parallel to these activities, a sub-ignition version FFHR-c1 ("before demo and compact"), where R_c= 13.0 m and V_p ~ 1,000 m^3, is under discussion to expand the strategic flexibility. Two options with different B_c are being considered. One is FFHR-c1.0 of B_c = 4.0 T, where NbTiTa superconductors can be selected, and the other is FFHR-c1.1 of B_c ~ 5.3 T, where the primary candidates are Nb_3Sn or Nb_3Al, and the counter option could be high-temperature superconductors. In FFHR-c1.0, P_f ~ 1 GW can be achieved with an auxiliary heating power of P_{aux} = 140 MW, *i.e.*, Q ~ 7. Then the maximum neutron flux onto the in-vessel components will exceed 10 dpa after one year operation. In FFHR-c1.1, on the other hand, the steady state sustainment of the plasma without P_{aux}, *i.e.*, self-ignition ($Q = \infty$), can be realized with P_f ~ 2 GW. A sub-ignition operation with a lower P_f is also possible. Although the full-power self-ignition operation should be limited to a short period less than a year, since the thickness of the shielding blanket is not as sufficient as FFHR-d1, the maximum neutron flux will reach ~10 dpa after 5 months. Therefore, FFHR-c1 can work as an efficient component test facility that can provide high-flux neutrons in steady state.

O8A.3 Biomass-Fusion Hybrid and Energy Application for Future Energy Strategy

Konishi, Satoshi[1]; Kasada, Ryuta

Kyoto University, Japan

Early fusion introduction with low Q reactor combined with energy multiplication system is considered as one of the possible strategy. This paper analyzes the energy application systems of fusion based on hybrid concepts, and describes technical issues for near term fusion energy plant as cross-cutting technology. Biomass-Fusion hybrid provides commercial fuels such as oil substitute or hydrogen that have socio-economic advantages. Plasma energy multiplication factor ~5 reactor similar to ITER can provide net positive energy output with assistance of biomass. This application utilizes endothermic reaction and essentially discards no waste heat. Carbon-free diesel can be produced from fusion energy combined with waste biomass. This product is competitive because few renewables can provide fuels, and its market is larger than electricity. Energy systems based on the realistic plasma, blanket and divertor technologies are designed and evaluated. Tritium control for the blanket and energy conversion system, and high grade heat extraction from divertor will be a challenge, and some possible solutions are presented.

Fusion electricity system to conform to the future grid is another issue. Adaptation of fusion to smaller grid and/or decentralized sources combined with renewables such as Photovoltaics and winds will be needed, for fusion introduction and deployment require stability of the entire grid. Impact of startup power of fusion devices on the grid is analyzed, and possible limit and electricity systems for future system with fusion is shown. Fuel production from fusion can be combined with a large numbers of dispersed fuel cell generators, that suggests another option of the fusion electricity. Finally, with such energy application systems, possible fusion introduction scenario in the global energy market is analyzed and shown. The result suggests that fusion can play a significant role in the world primary energy market in the latter half of this century with an innovative application.

O8A.4 Overview of engineering design, manufacturing and assembly of JT-60SA machine

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2. JT-60SA JA Home Team and 3JT-60SA Project Team, Japan Atomic Energy Agency, 801-1 Mukoyama, Naka, Ibaraki, 311-0193 Japan.

The JT-60SA experiment is one of the three projects to be undertaken in Japan as part of the Broader Approach Agreement, conducted jointly by Europe and Japan, and complementing the construction of ITER in Europe. The JT-60SA device is a fully superconducting tokamak capable of confining break-even equivalent deuterium plasmas with equilibria covering high plasma shaping with a low aspect ratio at a maximum plasma current of Ip=5.5 MA. This makes JT-60SA capable to support and complement ITER in all the major areas of fusion plasma development necessary to decide DEMO reactor construction.

After a complex start-up phase due to the necessity to carry out a re-baselining effort with the purpose to fit in the original budget while aiming to retain the machine mission, performance, and experimental flexibility, in 2009 detailed design could start. With the majority of time-critical industrial contracts in place, in 2012, it was possible to establish a credible time plan and, now, the project is progressing, on schedule towards the first plasma in March 2019.

After careful and focused R&D and qualification tests, the procurement of the major components and plants are now well advanced in manufacturing design and/or fabrication. In the meantime the dis-assembly of the JT60U machine has been completed and the engineering of JT-60SA assembly has been developed. The actual assembly of JT-60SA has started in January 2013 with the installation of the cryostat base.

The paper gives an overview of the present status of the engineering design, manufacturing and assembly of JT-60SA machine.

08:30 – 10:30
PARALLEL F2 (Topic C: Fuel Cycle)
Room 6

O8B.1 Estimate of Initial Tritium Inventory for the Fusion Nuclear Science Facility

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ITER is under construction and will operate in 2020. This is the first 500 MW class DT device, and since it is not going to breed tritium (with the exception of six test modules for the demonstration of tritium breeding), it will consume most of the limited supply of tritium resources in the world. With the estimated peak available tritium of about 28 kg around 2028 from heavy water reactors in Canada and Korea, after twenty years of ITER operation without breeding tritium the world supply of tritium will be only about 5 kg in 2040. After 20 years of operation, the accumulated neutron fluence of ITER will only be around 3 dpa. In order to generate a materials and fusion components database on exposure of up to ~100 dpa for DEMO, additional DT fusion component testing devices like Fusion Nuclear Science Facility (FNSF) will be needed. It is also reasonable to assume that an FNSF will be designed to generate net tritium at a tritium-breeding ratio higher than 1. But in order to get to the stage of net tritium production, it is necessary to supply the initial tritium inventory required for steady state DT operation, such that tritium equilibrium can be reached with an acceptable tritium generation doubling time. Our study assessed the initial tritium inventory for an FNSF-like-device by making estimates in three areas. The first area is the tritium inventory in the plasma stream for given plasma performance parameters, fueling efficiency, and wall recycling coefficient. The second area covers the tritium inventory in the tritium processing, extraction, cleanup and fueling systems. The third area covers the tritium inventory in the breeding blanket and corresponding systems, including inventories in the breeder material, neutron multiplier, coolant, piping, structural material and tritium extraction system. We are making a distinction between solid breeder and liquid breeder blanket options. The total estimated necessary initial tritium inventory for different steady state DT testing devices is about a few kg. This shows that the number of DT testing devices that the world can support will depend on the availability of tritium resources of the host organization or country. This paper presents estimates of the necessary equilibrium tritium inventories for FNSF-like devices in the range of 100–500 MW fusion power.

O8B.2 DT neutron irradiation experiment for tritium recovery on WCCB blanket

Ochiai, Kentaro[1]; Kawamura, Yoshinori[1]; Hoshino, Tsuyoshi[1]; Edao, Yuki[1]; Takakura, Kosuke[1]; Ohta, Masayuki[1]; Sato, Satoshi[1]; Konno, Chikara[1]

Japan Atomic Energy Agency, Tokai-mura, Naka-gun, Ibaraki, JAPAN

For the clarification of tritium recovery property of water cooled ceramic breeder (WCCB) blanket, we have performed a tritium recovery on-line experiment with DT neutrons at the Fusion Neutronics Source facility in Japan Atomic Energy Agency. Our previous experiment showed that the extracted tritiated water vapor (HTO) accounted for 90 % at 573 K. However, the previous one had no suppression of the humidity of H2O in the sweep gas line and then it was estimated that the water concentration was near 10000 ppm. Therefore, in order to suppress and monitor the humidity, we have added a molecular sieve and dew-point meter to the experimental sweep gas line and then have investigated the detailed recovery properties of HTO and tritium gas (HT).

The Li2TiO3 pebble of 70g was put into the stainless steel container which was set up it into an assembly stratified with beryllium and Li2TiO3 layer. The Li2TiO3 pebble was heated up to a constant temperature (573 K or 873 K). Helium sweep gas including H2 (1%) was flowed and extracted tritium was collected to water bubblers during DT neutron irradiation. The humidity of H2O in the sweep gas was held below 100 ppm with a molecular sieve. We also replaced the previous CuO bed by a more capacious one along the sweep line path to carry out separate measurement of HTO and HT.

From the present experiment, it was shown that the HTO extracted at 573 K was below 50 % of the total number of collected tritium and then H2O humidity in the sweep gas line made a difference to the ratio of HTO. It was also shown that the extracted tritium at 873 K was almost HT and the HTO was only about 5 %. It was found out that the ratio of HTO and HT was sensitive at the temperature of Li2TiO3 pebble.

O8B.3 The KALPUTREX-Process -A new vacuum pumping process for exhaust gases in fusion power plants

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At KIT Karlsruhe, a new vacuum pumping process is under development. This process, called the KALPUTREX-process, is based on a vapour diffusion pump as primary pump and a liquid ring pump as backing pump. As special feature, both pumps apply mercury as a liquid metal working fluid to make them tritium compatible. This pump combination can be built in a linear and theoretically arbitrary size (what allows also their application as NBI pumps) and fulfils high reliability, availability maintainability and inspectability (RAMI) requirements that are required for the economically attractive operation of fusion power plants.

Both pumps require a certain infrastructure that will also be presented in this paper: The diffusion pump needs a two stage baffle on the inlet side to avoid the backflow of working fluid into the vacuum chamber and, in addition, a heating and cooling system. This pump has no movable parts, is maintenance-free and will be powered by low-level heat that is extracted from the machine coolant to make it very energy efficient. The liquid ring pump needs a gas clean-up system that removes any mercury vapour out of the exhaust gas to protect the tritium plant.

In this paper, the KALPUTREX process is described in detail including the pumps and all infrastructure necessary for a safe operation of the machine vacuum system. Also, space requirements will be assessed and the implementation of the pumps in the machine infrastructure will be discussed.

O8B.4 Technologies and Modelling Issues for Tritium Processing in the European Tests Blanket Systems and Perspectives for DEMO

Ricapito, I.[1]; Calderoni, P.[1]; Poitevin, Y.[1]; Aieloo, A.[2]

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2. ENEA, Brasimone site, Bo, Italy

One of the main objectives of the experimental campaign of the Test Blanket Systems (TBS) in ITER is the demonstration of the efficient tritium processing and management from its source, the Test Blanket Module (TBM), up to its final routing to the ITER inner fuel cycle system.

On the other side, effective tritium processing and management in TBS have implication of basic importance on:

i) safe operation of TBS and ITER;

ii) development and validation of tritium transport modelling tools;

iii) demonstration of DEMO relevancy of some technologies. With respect to this last point, it has to be clarified that the selection among possible tritium processing technologies for the European TBS is driven more by compliancy with the safety and reliability ITER requirements than by DEMO relevancy aspects. However, testing in ITER some technologies foreseen in the current design TBS baseline like gas liquid contactors or vacuum permeators for tritium extraction for Pb-16Li, molecular sieve beds in the tritium extraction systems and, likely, tritium permeation barriers, is an important validation step towards possible applications to a DEMO breeding blanket.

This work describes various aspects of HCLL (Helium Cooled Lithium Lead) and HCPB (Helium Cooled Pebble Bed)-TBS activities that have a direct impact on TBS tritium management, with a focus on aspects important for support to DEMO Breeding Blanket design. In particular, it includes:

1. a description of the key tritium processing technologies and components in the current design baseline of the European TBS;
2. with reference to point 1), a discussion on the DEMO relevancy of some specific TBS tritium processing technologies;
3. an overview on the activities related to the tritium transport modelling tools that will be validated along the development of the TBM project, including experimental campaign in ITER, and used for supporting the DEMO Breeding Blanket design.

These three items are connected each other taking into consideration that tritium-related experimental data generated through the experimental campaign in ITER, interpreted through suitable modelling tools, will be one of the most significant outcomes in support of the breeding blanket design for DEMO and beyond.

08:30 – 10:30
PARALLEL F3 (Topic J: Plasma & Topic L: Fission-Fussion)
Room 5

O8C.2 Progress of integrated modeling for burning plasma control and operation

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Integrated modelling of tokamak plasmas is indispensable for predicting the performance of burning plasmas and developing control schemes and operation scenarios in ITER and future fusion reactors. It should describe MHD equilibrium, global stability, transport, and various external sources in a whole plasma including core, edge, SOL, and divertor regions, during whole operation period, including start up, sustainment, possible events, and shut down. Several initiatives aiming at the development of integrated modelling codes are under way in the world. The integrated tokamak modelling code TASK has been developed based on the Burning Plasma Simulation Initiative (BPSI), research collaboration of universities, NIFS and JAEA in Japan. It includes several levels of analysis, e.g. three levels of transport modeling: conventional diffusive transport, multi-species dynamic transport, and advanced kinetic transport. The last one is related to the kinetic integrated modeling [1] based on the momentum distribution functions of all plasma species. It has two main components, TASK/FP and TASK/WM. The bounce-averaged Fokker-Planck component TASK/FP [2] describes the time evolution of the momentum distribution functions, and the full wave component TASK/WM [3] describes not only ICRF wave heating but also Alfvén eigenmodes and low-frequency global eigenmodes. The Fokker-Planck component TASK/FP was applied to multi-scheme heating in ITER plasmas composed of electron, deuteron, triton and Helium ions. It was found that the energy dependence of the radial diffusion of energetic particles considerably affects the fusion output power. Finally in order to develop a real time control scheme of burning plasmas, a plasma simulator with external sources as inputs and diagnostics as outputs is required. It has to be verified by benchmark tests with other codes and validated by the comparison with experimental data. The integrated modeling code including diagnostics will be demonstrated as a prototype of such plasma simulator.

References

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[2] H. Nuga, A. Fukuyama, J. Plasma and Fusion Res. Series, 8, 1125 (2009)
[3] A. Fukuyama et al.: Proc. of 21th IAEA FEC (Chengdu, 2006) IAEA-CN-149/TH/P6-10.

O8C.3 Preparation for the operation of ITER: EU study of the pulse control system

Cavinato, Mario[1]; Ambrosino, Roberto[2]; Gribov, Yuri[3]; Mattei, Massimiliano[2]; Pironti, Alfredo[2]; Saibene, Gabriella[1]; Sartori, Roberta[1]; Zabeo, Luca[3]

1. Fusion for Energy, Barcelona, Spain
2. Association EURATOM-ENEA-CREATE, Italy
3. ITER Organization, France

In view of the preparation for the operation of the ITER tokamak it is necessary to develop the plasma scenarios taking into account all engineering constraints coming from the plant and including a realistic control system. It is important to consider that, due to the high energy of ITER plasmas, much more stringent requirements are posed on the control of transients in order to avoid machine damage.

Several activities are performed in the EU focusing on one side on the scenario optimization from a physics point of view and on the other side on the design and modelling of a realistic plasma control system driving the plasma configuration throughout the whole pulse and suitable for implementation on a the real machine.

The issues related to the computation of the control feed-forward component are addressed. In particular, the possibility to trigger a feed-forward component to solve controllability problems arising in the transitions from plasma L to H and H to L modes is studied in detail with the support of linear and non-linear simulations.

A control strategy is designed and tested on non-linear simulations of the whole pulse, including linear and non-linear effects due to controller switching, plasma shape reconstruction and power supplies.

The paper reports on the results of the studies performed and discuss the proposed design of the plasma control system.

O8C.4 Conceptual Design of a Fusion-Fission Hybrid Reactor for Spent Fuel Burning (FDS-SFB)

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Fusion-fission hybrid reactor is a practical path to the early fusion application for energy production, fuel breeding and waste transmutation since the fusion-fission hybrid has the potential attractiveness of easing the requirement of fusion plasma technology and plasma-facing material technology. The proposed roadmap and testing strategy of fusion application was presented based on the assessment and design analysis of the developed series hybrid system concepts in China as well as a summary of the hybrid system design and R&D activities in China and in the world by FDS Team.

In this contribution, a fusion-fission hybrid reactor for Spent Fuel Burning (FDS-SFB) which is a hybrid reactor concept for nuclear waste transmutation, fissile fuel breeding and energy production was proposed. It¡¯s designed based on relatively easy-achieved plasma parameters extrapolated from the successful operation of the Experimental Advanced Superconducting Tokamak (EAST) in China and other tokamaks in the world, and the subcritical blanket is designed based on the well-developed technologies of fission reactor. For FDS-SFB, besides the shielding modules, three types of functional blanket zones, i.e. tritium breeding zone, spent fuel burning zone and fissile fuel breeding zone are designed, respectively. For the inner board, it is only designated to breed tritium. FDS-SFB blanket makes use of spent nuclear fuel (SNF) and depleted uranium (DU) as fuel, a kind of RAFM steel named CLAM (China Low Activation Martensitic) as first wall and structure material, helium gas as coolant in the fission zone and liquid alloy LiPb as tritium breeder, nuclear fuel is adopted carbide formed with SiC cladding. The design and optimization of fusion plasma core parameters, blanket neutronics, blanket thermal-hydraulics, environmental impact analysis, tritium system and auxiliary system have been presented.

Keywords: Fusion, Blanket, Hybrid Reactor, Spent Fuel Burning.

11:00 – 13:00
PLENARY SESSION 3
Auditorium

I3.1 Overview of EU demo design activities

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Demonstrating the capability of generating several hundred MW of net electricity and operating with a closed fuel-cycle, in a DEMOnstration Fusion Reactor is viewed by many of the Nations engaged in the construction of ITER as the remaining crucial step towards the exploitation of fusion power after ITER.

ITER represents an indispensable step and significant advances in technology and physics are going to be acquired by its construction and operation. However, for a reactor, there are still some outstanding engineering challenges with potentially large gaps beyond ITER. A variety of fusion power system designs have been studied in the past across the world, but TRL analysis has often shown that the underlying physics and technology assumptions to be at an early stage of readiness.

One of the crucial points is the size of the device and the amount of power that can be reliably produced and controlled in it. This is subject of research and depends upon the assumptions that are made on the readiness of required advances in physics (e.g., the problem of the heat exhaust, choice of regime of operation, efficiency of non-inductive current drive systems, etc.), technology and materials developments. The recent EU fusion roadmap (Ref. [1]) advocates for a pragmatic approach and considers for the initial design integration studies a pulsed "low extrapolation" DEMO that could be delivered in the short to medium term. This should be based on mature technologies and reliable regimes of operation to be, as much as possible, extrapolated from the ITER experience, and on the use of materials adequate for the expected level of neutron fluence (see Ref. [2]). It argued that by waiting to design DEMO for the ultimate technical solutions in each area would postpone the realization of fusion indefinitely. However, it is clear that to realistically convert this outline concept into a reliable high performance facility there is a need for significant technical and science innovation. In addition DEMO must be capable of testing of advanced components and technical solutions that will be developed in parallel for application in a fusion power plant, thus playing the role of a component test facility as part of its mission.

This paper describes the progress on DEMO design and R&D activities in Europe by the EFDA Power Plant Physics and Technology (PPPT) where the emphasis is on the analysis of system requirements for a DEMO reactor, on the assessment of technical solutions and on the prioritization of R&D activities (e.g., power exhaust, power extraction and tritium breeding, RH schemes for high machine availability, qualification of radiation resilient structural materials, etc.).

A broad but integrated design-oriented approach is viewed as essential during this early concept design stage: (i) to better understand the problems and evaluate the technical risks of foreseeable technical solutions; (ii) to identify design trade-offs and constraints to address the most urgent issues in physics, technology and design integration; and (iii) to prioritize the R&D needs. Ensuring that R&D is focussed on resolving uncertainties in a timely manner and that learning from R&D is used to responsively adapt the technology strategy is crucial to the success of the programme.

Focus is now on the early implementation of the main technical activities identified in the fusion roadmap. Involvement of industry and exploitation of international collaborations on a number of critical technical aspects are desirable.

I3.2 Overview and status of ITER Internal Components

Merola, M.; Escourbiac, F.; Raffray, R.

The ITER Blanket and Tungsten Divertor Sections, and the procuring Domestic Agencies and Blanket Integrated Product Team

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The internal components of the ITER machine include the Blanket System and the Divertor. They directly face the fusionplasma and cover an area of about 850 m^2. They are cooled by pressurized water at 4.0 MPa and 70 C, inlet pressure and temperature, respectively. The main challenges of the internal components are: (1) the surface heat flux and neutronic volumetric heating, (2) the electromagnetic loads, (3) the various and sometimes conflicting project requirements (shielding, assembly, remote handling, integration with other in-vessel components, e.g. diagnostics, in-vessel coils, cabling, (4) high heat flux technologies and complex welded structures.

The main functions of the Blanket System are: (1) to exhaust most of the plasma thermal power, (2) to contribute in providing neutron shielding to vessel and ex-vessel components, including the superconducting coils and (3) to provide limiting surfaces that define the plasma boundary during startup and shutdown.

This system includes: (1) the First Wall panels, which are in front of the plasma and receive a surface heat flux up to about 5 MW/m^2, (2) the Shield Blocks, which support the First Wall panels, provide the neutron shielding, and, together with the First Wall panels, form the Blanket Modules, (3) the connections (flexible cartridges and pads), which mechanically attach and interface the Blanket Modules with the Vacuum Vessel, (4) the manifolds, which route the water coolant to the Blanket Modules.

A huge design and R&D effort has been carried out by the Blanket Integrated Product Team, which include the ITER organization and six Domestic Agencies (CN, EU, KO, JA, RF and US),and culminated with the Blanket Final Design Review in April 2013. The start of the procurement is planned by the end of 2013 with the signature of the first three Procurement Arrangements with CN and KO Domestic Agencies for the procurement of the Shield Blocks, and with RF Domestic Agency for the procurement of 40% of the First Wall panels.

The main function of the Divertor is minimizing the helium and impurity content in the plasma. The procurement of the present baseline, with a carbon and tungsten plasma-facing material (armour), was launched in 2010. However, in September 2011, the ITER Organization has proposed the possible start of the ITER operation with a full-tungsten armour Divertor in order to minimize costs and gaining operational experience with tungsten already during the non-active phase of the machine.

Since then, the design of this Divertor variant was initiated, and the already on-going physics and technological programme was accelerated. This substantial effort led to the full-tungsten Divertor Final Design Review in June 2013 and should enable to take a firm decision of the armour material by the end of 2013.

In parallel to the full-tungsten variant, the procurement of the carbon/ tungsten baseline was pursued with the manufacturing and cyclic testing up to 20 $MW/m^2$of a number of small and medium-scale mock-ups and with the start of the construction of full-scale prototypes in the three concerned Domestic Agencies (EU, JA and RF).

This paper summarizes the main requirements, the design and integration effort, the procurement, the related R&D and technology qualification of the Blanket System and the Divertorof the ITER machine and gives an outline of the concerned schedule.

I3.3 The JET technology program in support of ITER

Batistoni, P.[1,2]; Likonen, J.[3]; Bekris, N.[1]; Brezinsek, S.[4]; Horton, L.[1]; Matthews, G.[5]; Rube, M.l[6]; Sips, G.[1]; Syme, B.[5]; Widdowson, A. [5] and EFDA-JET Contributors*

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This paper presents an overview of the current and planned technology focused activities at JET in support of ITER operation and safety. The scope is very broad and ranges from detailed analysis of components from the ITER-like Wall (ILW) for material erosion, deposition and tritium retention to neutronics measurements and analyses for neutron detector calibration and code validation.

Following the first JET ILW operations in 2011-12, a range of post mortem technologies have been deployed from samples in remote areas that collect deposited material to special sub-surface marker layers on tungsten and beryllium tiles that allow precise determination of erosion rate. During operation, a dramatic reduction by one order of magnitude of C and of fuel retention has been measured by spectroscopy and by global gas balance, respectively. However, the fuel retention analysis in samples is critical for determining the true retained amount because the experimentally determined retention with the ILW during plasma operation using the active gas handling system is very low but can only be regarded as an upper limit due to long term outgassing. First results from ion beam analysis show a dramatic reduction in overall deposition as compared to the carbon wall and very low levels of retained deuterium with very little residual carbon. First results have also been obtained from special diagnostic mirror samples located in the main chamber and divertor which are critical for determining the design and the materials used for the first mirrors in ITER. In the future,

cleaning procedures for mirrors will be tested for these ITER-relevant deposits.
JET is the only fusion machine capable to produce significant neutron yields, typically ≈10^{16} n/s (2.5 MeV) in DD and up to nearly 10^{19} n/s (14.1MeV) in DT operations. Measurements of the neutron fluence, activation and shutdown dose rates are obtained inside JET and in the torus hall to validate the numerical predictions by the codes used in ITER design. In 2012, the absolute neutron fluence in the torus hall and in the wall labyrinth has been measured for the first time over about five order of magnitude variation range, and compared with numerical predictions.

Recently, the technological potential of a DT campaign at JET in support of ITER has been explored. The use of tritium with the ILW would provide accurate measurements to validate models of tritium retention, outgassing and removal efficiency in bulk materials, deposits and dust. Absolute calibration of neutron detectors using a D,T generator source will benchmark the ITER calibration procedure and assess the achievable accuracy. JET's unique 14 MeV neutron yield could be used to measure the activation of real ITER materials and the effect of irradiation on functional materials at doses in JET relevant to their location outside the ITER vacuum vessel. The neutronics codes and methodologies currently used in the ITER design would also be tested with the fusion relevant conditions in JET, with the appropriate neutron source and environmental complexities.

*See the Appendix of F. Romanelli et al., Proceedings of the 24th IAEA Fusion Energy Conference 2012, San Diego, US

POSTER SESSION 1
Tuesday 17th September
16:00 – 18:00
Ground Floor

Topic A First Wall

P1-001 **The New Method of Creation of Plasma and its Fast Heating till Thermonuclear Temperatures**
Chikvashvili, Ioseb

Individual

Modern TOKAMAKs provide sufficient for fusion confinement time (minutes order) and number density (several 1020 m-3) but due to low achieved temperature reactivity of plasma is still low. And even for ITER having projected Plasma Stored Energy 520MJ and total Heating Power 70MW, even neglecting inevitable losses, we would need not less than 7.4sec for temperature increase till achievement of self-sustained mode (ignition). And proposing by us Method unlike to using now others comprises in usage for plasma creation and its further ignition the created in-situ halo-layer of high-energetic particles to the puffed gas. For realization of Method the following procedures should be performed consistently: -to create time-dependent magnetic field penetrating only reactor's curvilinear segments -to apply toroidal magnetic field only in the regions located remotely from injection points -to inject 3 different kinds of pulse high current particle beams (two ions' – reacting components and one – electron's) with energies allowing them the capability of moving in a given bending magnetic field on a common equilibrium orbit in such a manner that faster ion beam passes through the moving at the same direction slower ion beam with sufficient for nuclear fusion collision energy and the relativistic electron beam moving oppositely to ions thus allowing to combined beam the self-focusing capability -to apply toroidal accelerating electric field As result of fusion the high energetic products can be produced, from which neutrons escape reactor while charged particles form halo-layer. Then, at once after injection: -from the walls with the help of corresponding valves to puff the gas consisting the main fuel components. That gas will absorb halo-layer's energy thus being ionized and heated -to apply toroidal magnetic field at injection points too The Method allows the reliable ignition of plasma in all kinds of toroidal fusion reactors.

P1-003 Fabrication of W/FMS joint mock-ups for first-wall using a hot isostatic pressing

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Tungsten is attracting a lot of interest as an armor material for the future nuclear fusion power plant owing to its high wear resistant, low tritium retention, and reduced activation properties. Joining of W to steels and copper alloys and steels has been investigated in order to develop the manufacturing technologies for plasma facing components. It is difficult to fabricate W/steel or W/Cu joints because of the large difference in their thermal expansion. In this study, W/FMS joint mock-ups were fabricated without any interfacial defects or cracks. For the joining, double-staged hot isostatic pressing (HIP) was applied. The first stage HIP (900 ℃, 100 MPa, 1.5 h) was aimed at forming diffusion joining interfaces between W and FMS. The second stage HIP (750 ℃, 70 MPa, 2 h) corresponds to a tempering process to retain the mechanical properties of FMS. The tempering was performed through a HIP process in order to suppress the edge delamination of W/FMS joints during thermal history. As an interlayer material, a titanium foil that can mitigate the thermal expansion difference between W and FMS was inserted. In addition, a molybdenum foil was introduced to prevent an unwanted bonding of W to a canning material. It was found that the stainless steel 304L (i.e., canning material) is easily form a diffusion bonding with W, which caused the shear stress in W plates as well as at W/FMS interfaces during the thermal process. The lateral cracks in W plates (usually observed in the case of a conventional HIP process) were not observed when the molybdenum separator was used. According to current HIP process, W/FMS joint mock-ups with the dimension of 50 x 50 x 32(t) mm were fabricated. The mock-ups were absent of any interfacial cracks and defects. The shear strength of the joints was 82 MPa in average.

P1-004 **Design Strategy for the PFC in DEMO Reactor**

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The life-time performance of functional and structural materials in fusion reactor DEMO is limited by several processes such as a neutron damage, plasma erosion and transient events. The design strategy of the plasma facing components in DEMO reactor is to determine the structure and coating thicknesses which maximize component lifetime against all life-time limitations. A sandwich type first wall blanket module made of W-clad EUROFER steel is examined against the normal and off-normal operation heat loads expected in DEMO. A feasibility of the conventional finger type W divertor plate module is examined against the power loads and consequent erosion caused by plasma impact during transients and off-normal events.

It is shown that apart from the fact that W/EUROFER bound is compatible with high neutron fluencies and is "low-activation" type (thus minimizing the necessary replacement of the in-vessel components), EUROFER steel as a structural material will remain creep resistant in the case of the 'hot wall' operation. However, high temperature wall causes a fundamentally different physical chemistry regime for wall surface erosion. After calculation of erosion and thermal material destruction due to plasma impact by means of MEMOS and ENDEP codes (B. Bazylev et.al, J. Nucl. Mat. 307-311, 69 2002), we present the lifetime analysis for W and EUROFER materials under DEMO operation conditions. Finally we consider the efficiency of helium and supercritical water as a coolant and compare their advantages and disadvantages for high temperature DEMO operation.

P1-005 **Numerical study of the impact of hydrogen bombardment on the mechanical properties of tungsten**

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Tungsten has been considered as the most promising plasma facing material for the next fusion devices because of its lower sputtering erosion and good thermal properties. For ITER, tungsten will be used as divertor armor, where it's supposed to withstand high flux bombardment of hydrogen isotope ions and intense heat loads which could introduce high thermo stresses and therefore lead to the failure of the material. In this work, molecular dynamics simulations were performed to investigate the mechanical behavior of tungsten with and without hydrogen atoms under uniaxial tensile load. The stress-strain curves for different hydrogen concentration were obtained. It was found that hydrogen has great influence on the elastic modulus and deformation mechanism. By simulating and analyzing the interaction between hydrogen atoms and dislocations, we believe that hydrogen atoms could weaken the bonding between tungsten atoms and therefore increase the mobility of dislocations, which leads to the macroscopic softening. Furthermore, the qualitative relationship between yield stress and hydrogen concentration was given, which confirms the softening effect of hydrogen.

P1-007 **Joining of HHF components applying electroplating technology**

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Tungsten will be used as armour material for blanket shielding and is designated as high heat flux material for divertors, beyond application of improved W composite alloys as structural material. Independent from design (water- or helium-cooled), a successful divertor development is inherently correlated with joining of tungsten with functional components, fabricated from a variety of alloys ranging from Cu to steel, e.g. Eurofer to advanced W grades / composites. Depending on the design variants, the fabricated joints have to guarantee specific functional or structural properties, e.g. good thermal conductivity or mechanical load transmission. Tungsten shows lacks in adapted joining due to its metallurgical behavior ranging from immiscibility over bad wetting, brittle intermetallic phase formation up to large expansion mismatch. Overcoming these drawbacks requires development of improved and adapted joining technologies. Electroplating has shown that homogeneous functional and filler layers can be deposited on tungsten which led to encouraging progress in wetting. Functional Ni interlayers were selected to act as model alloy on the development path towards more fusion favored metals like Pd or Fe which are under investigation for joint fabrication. In this paper the progress achieved in development of electroplating processes for joining W to W or W to steel will be shown. Development of Pd-Cu alloy deposition by electroplating as first interlayer on W will be outlined, too, concerning deposition under process stability considerations in comparison to Ni-Cu. The applicability of electroplating in general as appropriate joining technology will be shown by presenting metallurgical testing and SEM/EDX analyses of fabricated joints. The mechanical stability of the produced joints is demonstrated by presenting shear test data. Further application areas, e.g. development of W composites will be outlined, too.

P1-008 **Er2O3 coating: process optimization through film characterization**

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We develop an experimental set-up for high temperature reactive magnetron deposition of Er2O3 (Erbia) – the functional coating to mitigate tritium permeation into the containing walls of fusion reactor breeder. It is known that electric resistivity of Er2O3 is sufficiently high to tackle the MHD drag problem in liquid metal cooled breeder design. Indian proposed test module (Lead-Lithium cooled Ceramic Breeder) for ITER may require dual function of tritium barrier and electrical insulation to avoid MHD drag, of such coating, unlike the requirement in liquid breeder and solid breeder concepts. So we aim to develop Er2O3 coating with desired properties. Here, we present our results on optimization of the coating process on different parameters. The optimization is aimed at forming Er2O3 coatings in cubic phase, which is the most stable type of its structures. Additionally, we are also concerned to get such coating at reasonably good deposition rate. We also explore if crystalline orientation of such coating can be controlled by tuning the parameters in reproducible manner. To achieve these, we systematically deposited the coating by varying parameters such as sputtering power, substrate temperature, annealing in O2 rich vacuum, O2 content during sputtering process, etc. The most important of our results suggest that we can get cubic Er2O3 at a significantly lower temperature of 360 C as compared to the reported results, by controlling O2 content precisely while keeping all other parameters constant at optimized values. Importantly, we also get very good deposition rate of 12 nm/min and higher. The other crystalline phase observed is monoclinic, especially at lower temperature and/or at higher O2 content. Various optimization experiments invariably resulted in Er2O3 phase, either in cubic, monoclinic or amorphous phase as inferred from the X-ray diffraction, SEM cross-section and visually pinkish and transparent nature of the coating.

P1-009 **Recent Advances of T-11m Lithium Program**

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Previous experiments with limiters with lithium PFC on the basis of thin capillary-porous structures (CPS) in T-11M and FTU tokamaks have shown promising results: significant decreasing Zeff(0) in the center of plasma column due to lithization and decreasing heat loading on lithium surface PFC of limiter due to reradiation by lithium in the periphery of plasma column. At present, lithium T-11M program is focused mainly on the tasks of a steady-state tokamak: creation of an emitter-collector limiter system in the tokamak SOL in oder to protect PFC of tokamak by Li and to minimize the amount of lithium in the tokamak vessel. Recent lithium activity in T-11M has had three directions: investigation of different Li limiters and a new kind of CPS lithium PFC in tokamak conditions, investigation of Li migration and circulation in SOL region, research and development of methods of extracting lithium from the tokamak vessel without depressurization. In all previous experiments rail limiters were installed horizontally at the bottom of the tokamak vessel. The size of the horizontal port allows to install a longer rail limiter that can reduce losses of lithium on the walls of the tokamak vessel. We present the results obtained in experiments with vertical lithium limiter upright in the T-11M tokamak (R=0.7m, a=18÷20cm, Ip=60÷100kA). Radial profiles of ion fluxes of deuterium and lithium in the SOL were investigated for various plasma densities. It is shown that distributions have the exponential form with e-fold lenght λ = 3.5cm for lithium and λ = 5.2cm for deuterium plasma. Estimations of circulation of the part of lithium, returning to the surface of the vertical lithium limiter according to the radial profile give the value of about 75%. A new cryogenic collector has been tested for lithium removal in GD from tokamak vessel wall. The collection efficiency of lithium was at about 1 mg per hour, which corresponds to the collection of lithium deposit on the wall during 50 ordinary shots of T-11M or two days of work.

P1-010 **Helium-implanted CLAM steel and their annealing behavior investigated by positron-annihilation spectroscopy**

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China Low Activation Martensitic (CLAM) steel has been chosen as the primary candidate structural material for the first wall/blanket for fusion reactor. The Helium irradiation induced damage of CLAM steel and the annealing behavior of defect were investigated in this paper. All the samples were annealed at 673K for 2.5 hours in high vacuum atmosphere before irradiation. Post-implantation annealing behavior was investigated for CLAM steel, which was homogeneously implanted with1e+17ions/cm2 of Helium at room temperature, 473K, 673K and 873K. The He2+ energy is 100 keV. Irradiation induced damage of CLAM steel and the annealing behavior of defects were probed by slow positron beam Doppler broadening technique. Helium implantation produced a large number of vacancy-type defects in CLAM steel, and small vacancy clusters and Helium bubbles were the dominant defects. It can be obviously observed that there exists a decrease of vacancy in the annealed samples. Below 673K, the S parameter increased with increasing temperature and Vacancy-Helium complexes were main defects after Helium implantation, both the formation and decomposition of vacancies existed. Target atoms displacement capacity was stronger than the annihilation of vacancies. The S parameter decreased at 873K, Helium bubbles would be unstable and the desorption of Helium atoms would promote the annihilation of vacancies. The annihilation became significant with increasing temperature, therefore, the amount of vacancies reduced leading to the decreasing of the S parameter.

P1-011 **Manufacturing of ITER pre-qualification Normal Heat Flux (NHF) First Wall (FW) 2 MW/m2 small-scale mock-ups and semi-prototype**

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This paper describes the manufacturing development and fabrication of reduced scale ITER First Wall (FW) mock-ups of the Normal Heat Flux (NHF) design, including a 'semi- prototype' with a dimension of 305 x 660 mm2, corresponding to about 1/6 of a full scale panel. The activity was carried out in the framework of the pre-qualification of the European Domestic Agency for the supply of the European share of the ITER first wall. The hardware consists of three Upgraded (2 MW/m2) Normal Heat Flux (U-NHF) small-scale mock-ups, bearing 3 beryllium tiles each, and of one semi-prototype, representing six full-scale fingers and bearing overall 84 beryllium tiles. The manufacturing process makes extensive use of Hot Isostatic Pressing, which was developed during more than a decade during ITER Engineering Design Activity phase. The main manufacturing steps for the semi-prototype are described, with special reference to the lessons learned and the implications impacting the future fabrication of the full-scale prototype and the series which consists of 218 panels plus spares. In addition, a 'tile-size' mock-up was manufactured in order to assess the performance of larger tiles. The use of larger tiles would be highly beneficial since it would allow a significant reduction of the panel assembly time.

P1-012 **Three-dimensional flow measurement of a sphere-packed pipe by a digital hologram and refractive index-matching method**

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Moltensalt as coolant material in fusion reactor have a significant role in the design of advanced reactors. The investigation of thermal behavior in an actual facility environment is needed with heat transfer enhancement under high Pr number fluid flow such as FLibe. It is important for the development of the facility to detect fluid motion of a basic heat transfer promoter such as a sphere-packed pipe (SPP). In the present study, digital holographic PTV visualization to identify the complex flow structures in a sphere-packed pipe is carried out by using a refractive index-matching method with a sodium iodide (NaI) solution as a working fluid. This solution is deliberately chosen to be able to adjust the refractive index of the working fluid to that of the acrylic sphere with an index of 1.49. An observation region is made by an acrylic acid resin box which hollows out a cylindrical domain of 16mm diameter. A sphere-packed pipe (SPP) was created; the acrylic sphere with a diameter of 8 mm is packed in the observation region. This technique enables a clear view through the sphere. When the refractive indices between the working fluid and the sphere were matched, the sphere is quite invisible to the eye. The hologram fringe image of the particles behind the sphere can be observed and the particle positions can be reconstructed by a digital hologram. Consequently, three-dimensional velocity field around the sphere is obtained by the reconstructed particles position. The velocity near the pipe wall region is faster than that in the central pipe region, that is, the velocity in the region can contribute the thermal enhancement in thin thermal layer such as high Pr number fluid flow.

P1-014 **Analyses results of the EHF FW Panel with welded fingers**

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According to Blanket Procurement Arrangement Simplification (PCR-244) about 40% of the FW panels (176 Enhanced Heat Flux panels) shall be delivered to ITER by Russian Federation Domestic Agency (RF DA). There are two RF institutions are involved in the development and manufacturing of the EHF FW panels: D.V. Efremov Scientific Research Institute of Electrophysical Apparatus (NIIEFA) and N.A.Dollezhal Research and Development Institute of Power Engineering (NIKIET). In 2012 RF DA specialists have been developed a new design of EHF FW panel. The main innovation is as follows: two welds are used for regular fingers fixation on FW beam instead of pinned joints. In order to provide necessary thickness of welds the redistribution of material between finger steel body and FW beam has been done. Also the 60 mm width slot has been performed in the middle part of FW panel in order to provide the access for welding equipment. As a result the problem of protecting the central slot from plasma radiation has been appeared. RF DA specialists propose to use special insert for this purpose. The design of this insert is look like as regular finger design: beryllium armour, CuCrZr heat sink layer and steel base. It is should be noted that hydraulic scheme has been developed for new design of EHF FW panel and central insert coolant path has been successfully integrated in it. This article summaries some results of thermo-hydraulic and combined structural analyses of EHF FW panel and it's components for Inductive I operation mode.

P1-015 **Production Management and Quality Assurance for the Fabrication of the In-Vessel Components of the Stellarator Wendelstein 7-X**

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The In-Vessel Components (IVC) of the stellarator Wendelstein 7-X consist of the divertor components and the first wall (FW) with their internal water cooling supply. Due to the significant amount of different components including many variants a tool called Production Managing System (PMS) has been used to organize the fabrication and the associated quality assurance. The PMS operation relies on the building of a database which contains the basic part and assembly data, working and quality control plans, and working machine capacity. The creation of this database based mainly on the parts lists, the manufacturing drawings, and details of the working flow organization. Owing to the learning process and technical adjustment during the design and manufacturing phase, the database has been permanently updated. In consequence of this, an interface to optimize the data preparation has been developed. PMS has been demonstrated to be an efficient tool to support the IVC activities with a good prediction of the planning, a fast adaptation and an efficient re-organization possibility due to encountered problems during the fabrication or change in the project priorities with the associated milestones. This paper presents the database design and installation, the quality assurance plan, and its application to the fabrication of IVC.

P1-016 **Analysis and Primary Experiment Results of a Guidable Free Curve-Surface Flow for Liquid Metal PFCs**

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Liquid metal (LM) free surface for LM divertor/limiter have been studied in a widely range in the last decade, such as film flow, jet flow, capillary and so on. But what kind of free surface flow can be suitable to LM plasma facing components (PFCs) and what MHD effect will be caused? It is key issues to be studied. An innovation concept of multi-layers guidable liquid metal free curve-surface flow is addressed and its establishment & magneto-hydrodynamics (MHD) behaviors in a gradient magnetic field are also studied on theoretically and experimentally. The free curve surface is composed by three layer flows. Layer I is a basic conduction layer, layer II is a key adjust layer, layer III is the surface layer. The analysis results show that an MHD effect stability free curve surface flow (Layer III) can be established under a given curve surface by adjusting the gap between the plate and meshes and the frication coefficient of meshes under a fixed magnetic field profile and initiation velocity of the flow. The experimental result indicates that the curve plate-multilayer meshes system, which was built base on the theoretical results, can be efficiently conducting the free curve surface flow to avoid rivulet flow, and that the transverse gradient magnetic field made free curve surface flow more stability and smoothly surface. Moreover, in the curve plate without mesh case, the transverse magnetic field enhanced the secondary flow effect.

The experimental parameters are the magnetic field at 1.20 -1.85 Tesla, frees curve surface flow initiation velocity at 1.2 m/s, the flow 50 mm in width, about 10 mm in thickness, 600 mm in length.

P1-017 **Mechanical Analysis of the Joint between Wendelstein 7-X Target Element and the Divertor Frame Structure**

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For the long pulse operation of Wendelstein 7-X (W7-X) it is foreseen to have an actively cooled high heat flux (HHF) divertor [1], which consists of individual target modules. Each module is a set of target elements made of CFC tiles bonded to a CuCrZr heat sink mounted on to a support frame. During operation, the power loading of the element will result in the thermal expansion of the target elements. The attachment to the support frame needs to provide, on the one hand, enough flexibility to allow some movement in order to release the induced thermal stress and, on the other hand, the mount has to provide enough stiffness to avoid a misalignment of one target element relatively to the others, which could cause a leading edge phenomena. This flexibility is realized by a spring element; build up from a stack of disc springs together with a sliding support at one of the mounting points. Detailed finite element calculations have been performed including the nonlinear behavior of the spring elements to evaluate the deformation under the specified stationary load of 10MW/m². The results of the calculations showed that the deformation of the cooling structure leads to some non-axial deformation of the spring elements. Thus a mechanical test bed was built to qualify the stability of the attachment under calculated cyclic loading and deformations typical of the expected deformation of the elements.

The paper describes the tests performed, the experimental results of the strain gauges and the bolt pre-tension under cyclic loading and compares them to the finite element calculations.

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P1-018 **Effects of heat treatments on deuterium retention/desorption properties of tungsten materials**

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Tungsten (W) is a leading candidate of the first wall material in the fusion reactor. Hydrogen isotope retention in the first wall is a key concern associated with safety and density control in the reactors. The effects of impurity and microstructure of W material on its hydrogen isotope retention/desorption is still unclear. In the present study, deuterium were implanted to various W materials, then the deuteirum retention/desorption behaviors were investigated. The behaviors of W with various heat treatments were also evaluated. The effects of impurity and the surface structures of the W materials on the D retention/desorption were discussed. TiC-mixed W, and W with stress relieving process at 1173K (Stress-relieved W) or with recrystallizing process at 1773K (Recrystallized W) were used as samples. In addition, Stress-relieved and Recrystallized W with additional heat treatments at 1273K were also used. The sample was irradiated with deuterium ions with energy of 1.7 keV at RT. After the irradiation, the deuterium retention/desorption behaviors were evaluated by thermal desorption spectroscopy. In addition, the surface morphologies and microstructure before and after the irradiations were observed. After the irradiation, the blisters on the surface of Stress-relieved and Recrystallized W were observed. The blister was hardly formed after the additional heat treatment. While there were also the effects of C on the blister formation, there are no significant difference between the retention/desorption behaviors of pure and TiC-mixed W. Stress-relieved W had a characteristic desorption peak, which might be associated with C impurities. This peak disappeared and the retention was decreased by the additional heat treatment. The results indicate the heat treatment affect the impurity contents and the microstructure, which lead to change in its retention/desorption properties. These results suggest that there was the possibility of the change in the hydrogen isotope retention of W during the reactor operation.

P1-019 **Simulation of neutral gas flow in a tokamak divertor using Direct Simulation Monte Carlo method**

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Divertor design and optimization have become one of the most challenging subjects in the fusion community. The first difficulty relies on describing accurately the physical processes between plasma, neutral particles and the divertor surface. For ITER, the physics and optimization of the divertor have been tackled via SOLPS (formerly B2-EIRENE) code [1, 2, 3]. However, future fusion reactor conditions like DEMO envisage higher neutron loads [4, 5], higher helium exhaust and significant amount of impurities from the first wall. These loads and impurities are then pumped so as to sustain long-pulse operation. Thus, from the perspective of the pumping system it is crucial to study in detail the flow behavior in different operational conditions of the reactor and how this flow affects the efficiency of the related vacuum system. In the present work, numerical simulations of the flow field and overall quantities of practical interest inside the ITER's 2008 divertor are done by coupling the SOLPS package and the Direct Simulation Monte Carlo (DSMC) algorithm. On the one hand, the present study unravels the gas recirculation effect and its influence on the vacuum system performance and on the other hand this approach provides an innovative scientific and engineering tool for design and optimization of the divertor system for future fusion devices.

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P1-020 **Simulation of runaway electron evolution during a disruption in HL-2A tokamak***

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Runaway electrons (REs), occurring during the plasma disruptions, usually carry huge heat load and pose a severe threat to the plasma facing materials (PFMs) of fusion devices. Since the appropriate diagnostics for REs during disruptions are scarce, theoretical modeling is an indispensable means to understand the generation mechanisms and evolution of REs. In this work, the RE phenomenon observed during a major disruption in the HL-2A tokamak[1] is simulated with a self-consistent model[2], which is constituted by Maxwell¡¯s equation and generation rate equation. The generation rate equation includes Dreicer and avalanche generation mechanisms of REs. Most of the input parameters for the simulation are taken from the HL-2A experimental database (discharge No. 15335). The simulation results show that REs are produced mostly in the region around the magnetic axis and the post-disruption current is maintained stably by REs for tens of milliseconds. The simulation also shows that the electric field increases rapidly from 0.15 ms after the disruption, reaches the maximum about 0.15 ms later, and lasts about 2.5 ms. Moreover, it is found that 42% of the Ohmic current is converted into runaway current in HL-2A, which is in good agreement with the experimental value 55% [1], and that the avalanche generation of REs plays a minor role in HL-2A, which is consistent with the small magnitude of observed plasma current [1]. In addition, the toroidal effects, the influence of different   values and the radial diffusion of REs are further discussed in the paper.

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P1-021 **Molecular dynamics simulation of the energy deposition of low energy hydrogen and its isotopes in tungsten**

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Dust particles, resulting from plasma-wall interactions in modern magnetic fusion devices, pose a critical issue to tokamak operation and safety. Dust may be radioactive, and contain toxic substances from the vessel wall. It may penetrate into the core plasma and even occasionally lead to the termination of plasma discharges [1]. Therefore, it is important to investigate the dust transport and its damage to the wall in fusion devices. For this purpose, we have developed a dust transport simulation (DTS) code to analyze recent experimental results on dust formation and basic characteristics in EAST [2]. The DTS code contains four modules [3] to model dust charging process, heating and ablation, transport and its interactions with the wall. The charging mechanism takes into account the charge contribution from ions, electrons, impurities, thermionic currents and the second electron emission. The heating process includes energy absorption from impinging particles, recombination of plasma ions on dust surface and radiation. The dynamic charges, sizes and masses of dust particles obtained from the above processes are then transferred to the transport section at every time step. To model dust dynamics in the realistic tokamak geometry, the SOLPS code is utilized to provide the background plasma information such as plasma density, temperature and electromagnetic field. The coupled DTS-SOLPS code package offers an efficient tool for the integrated modeling of dust in magnetic fusion devices. This paper presents, for the first time, the numerical simulation of dust in EAST using the newly developed DTS code, coupled with SOLPS. The code can track the dust particles produced at different locations in EAST and quantify their damages to the chamber wall. Detailed information has been obtained on the initial size and location of dust, its velocities, trajectories and resultant wall damages at different components etc. has been obtained. This should provide useful information for preventing dust formation and its contamination to the core plasma, which is essential for high power long pulse operation of EAST.

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P1-022 **Evaluation of heat transfer by sublimation for the application to the divertor heat sink for high fusion energy conversion**

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For fusion power reactor such as DEMO, divertor is required to not only remove high heat flux, but also to transfer it to high temperature coolant media for the efficient utilization of energy. The authors have proposed a new concept of divertor heat sink with phase transition of liquid metal. This study evaluates the heat transfer capability from the localized heat load on the divertor to a coolant for heat exchange with a small temperature difference, and analyzes its possible limits as the functions of geometry and operating temperature. Taking advantage of the similarity with those of heat pipe, sonic limit and entrainment limit in heat pipes were considered. For the application to the divertor heat sink, formulae for the heat pipes were modified to incorporate vapor flow space and liquid path for working fluids to be independently designed. In case of water heat transport system with heat transport capacity on 10cm2 area is expected to exceed 10MW/m2 at 97 degree C limited by the vapor flow. In case of Na heat transport system, heat transport capacity on 10cm2 area is also expected to exceed 10MW/m2 at 727 degree C, however entrainment limit was more dominant than sonic limit in the considered temperature range. The result suggested that the divertor heat sink based on this dual phase heat transfer system should be designed to have balanced liquid and vapor flow paths at the desired operating temperature. Experimental verification is compared with this analysis and the design of the divertor is made based on it.

P1-023 **Prototyping of the Blanket Shield Module for the ITER ECH&CD Upper Launcher**

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The ITER Electron Cyclotron Heating and Current Drive (ECH&CD) Upper Launcher is recently in the first of two final design phases. The first phase deals with the finalization of all FCS (First Confinement System) components as well as with specific design progress for the remaining In-Vessel components.

The most outstanding structural In-vessel component of an ECH&CD Upper Launcher is the Blanket Shield Module (BSM) with the First Wall Panel (FWP). The both of them form the Plasma facing part of the launcher, which has to meet strong demands on dissipation of nuclear heat loads and mechanical rigidity. Nuclear heat loads from 3 W/cm³ at the First Wall Panel´s surface, decaying down to a tenth in a distance of 0.5 meter behind of it will affect the BSM and the FWP. Additional heating of maximum 0.5 MW/m² due to plasma radiation must be released from the FWP.

To guarantee save and homogenous removal of such extensive heat loads, the BSM is designed as a welded steel-case with specific cooling channels inside its wall structure. Attached to its face side is the FWP with a high-power cooling structure.

Based on computational analysis the optimum cooling channel geometry has been investigated. Specific pre-prototype tests have been made and associated assembly parameters have been determined in order to identify optimum manufacturing processes and joining techniques, which guarantee a robust design with maximum geometrical accuracy. This paper describes the design, manufacturing and testing of a full-size mock-up of the BSM. The study was carried out in an industrial cooperation with MAN Diesel and Turbo.

P1-024 **Infrared thermography inspection for mono-block divertor target in JT-60SA**

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Carbon fiber Composite mono-block divertor target is required for power handling in JT-60SA. For keeping a heat removal capability of the divertor in JT-60SA, a jointing technology and quality control of the target are key issues in the manufacturing process. Therefore infrared thermography inspection (IR inspection) proposed from SATIR is modified to improve an accuracy of a screening for the heat removal capability. In the IR inspection, the targets heated at 95 oC by hot water in steady state condition are instantaneously cooled down by cold water flow of 5 oC in three channels of test section. The heat removal capability of the targets is evaluated with comparing the difference of the transient thermal response between defect-free and tested targets. Because a screw-tube, which has a significantly short time constant of cooling the target, is used as cooling tube in the target of JT-60SA, difference of starting time for switching from the hot water to the cold water between each channel is not negligible. To overcome this problem, defect size is evaluated with the cooling time from 90�‹C to 60�‹C in each target. The IR inspection and a heat load test with an electron beam irradiation are performed for mock-ups with an artificial defect in an interface between the mono-block and the cooling tube to establish criteria of the screening. Moreover, an adequacy of the inspection is assessed by comparing the results with the finite element method. In this study, a construction of a database for a correlation between the known defects, maximum surface temperatures in the heat load test and the transient thermal responses in IR inspection are successfully completed. Moreover, the criteria are evaluated from the kinds of the defects and a magnitude of the heat load.

P1-025 **Study on Deuterium Retention and Lithiation Properties of Tungsten**

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High Z materials like tungsten have been considered to be good candidates for the plasma facing material in the divertor region of ITER. Lithiation can significantly improve plasma performance in long pulse tokamak, like EAST. The investigation of lithiated-tungsten is important for understanding lithium conditioning effects in EAST, where tungsten will be used as plasma facing material. In this presentation a few important issues of lithiated-tungsten will be studied, such as, the chemical state of lithium on the tungsten substrate, the effect of lithiation on deuterium retention and the profile of element distributions on the lithiation layer. Deuterium retention effects in pure tungsten and lithiated-tungsten have been investigated on the MFCAP-PSI (Dalian) and MAGNUM-PSI (DIFFER) linear plasma simulators. Laser-induced Breakdown Spectroscopy (LIBS), Scanning Electron Microscope (SEM), X-ray Photoelectron Spectroscopy (XPS) and profilometer techniques were used to measure the D retention and physical and chemical properties of the lithiated tungsten. The results indicate that, after D plasma exposure, D retention could be saturated in the lithiation layer and Li in the lithiated-layer has strong chemical bonds with O, H, C, but no chemical bond with tungsten. The results could be useful for applying LIBS as a wall-diagnostic technique for EAST.

P1-026 **Dual-pulse laser induced breakdown spectroscopy for measuring laser cleaning process of co-deposition layer on the first mirror of HL-2A**

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Our recent investigations have indicated that dual-pulse laser-induced breakdown spectroscopy (LIBS) may have great potential in the on-line measurement of the laser ablation process of the ultrathin co-deposition layer on the first mirrors of HL-2A tokamak during the cleaning. The spectroscopic study of the plasma emission was used to determine the elemental composition of the co-deposition layer. The on-line implementation of the spectroscopic technique LIBS to the cleaning process provides important information about the optimal cleaning parameters in order to avoid over-cleaning. The results indicate that the co-deposition layer (about 800 nm) is almost completely removed after multiple pulses at about 0.5 J/cm2, but the plasma initiated by the cleaning laser pulse with the low-energy density is too weak to be used for producing LIBS signals. To overcome this problem, dual-pulse LIBS has been proposed for the first time. In this approach the cleaning laser irradiates the polluted mirror at first, which is used to ablate the deposits and to form weak plasma. A few microseconds later, the second laser pulse propagating parallelly to the mirror surface and focusing to the weak plasma is applied for re-heating. In this way, an enhanced LIBS signal was observed. Some details will be discussed in the meeting.

Topic B Blankets

P1-027 **Preparation of Al2O3/YSZ Multi-laminated Coatings by Sol-gel Technique as Tritium Permeation Barrier**

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Covering permeation barrier on structural materials to mitigate tritium transfer from Pb-Li into the water/helium coolant is a feasible solution in test blanket module (TBM) system. Alumina is one of the most promising coating materials, due to excellent stability and barrier effects. However, thermal mismatch between alumina and martensitic/ferritic steel would shorten its service life, especailly in the condition of temperature fluctuation. From this viewpoint, a material with bigger coeffecent of thermal expansion (CTE) is more suitable to prepare tritium permeation barrier. Our previous study [1] indicated that zirconia is a potential candidate for this purpose. In this study, Al2O3/YSZ multi-laminated coatings was prepared by sol-gel technique as tritium permeation barrier. X-ray diffraction (XRD) and field-emission scanning electron microscope (FSEM) were employed to identify the phase and examine the microstructure of the coating. The thickness of the coating can be controlled by the number of dipping cycles. The flexibility to coat complex geometries by this method, even inside tubes, is guaranteed.

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P1-028 **Studies on the solubility of hydrogen in molten Pb83Li17 eutectic alloy**

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Lead-lithium eutectic (Pb87Li17) alloy is a candidate material to be used as a secondary tritium breeder, neutron multiplier and heat transfer agent in the fusion reactor. The tritium thus produced in the alloy may be soluble or appear as a new phase of lithiumtritides and/or lead-tritides, which eventually affect the performance of Pb83Li17 eutectic. Therefore, solubility of tritium in the alloy at the operating conditions of the fusion reactor is a subject matter of investigation. Tritium being the isotope of hydrogen behaves more or less similar to the hydrogen. In the present investigation the solubility of hydrogen in the Pb83Li17 has been investigated as a function of temperature and pressure. It was found that, hydrogen solubility in the Pb87Li17 alloy is almost constant above 350ºC. Hydrogen solubility increases with increase of temperature up to 400ºC. Hydrogen solubility is 120 ppm at 400ºC and 800 Torr hydrogen pressure. The solubility decreases on further rise in temperature from 400ºC. However, at all the temperatures hydrogen solubility increased with increase of partial pressure of hydrogen.

P1-029 **Results of LLCB TBM conceptual design optimization**

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Breeding zone optimization and choice of testing parameters in support of Indian conceptual design of lead-lithium ceramic breeder test blanket module (LLCB TBM) for ITER are presented. LLCB TBM is of dual coolant type: LL in the breeding zone and helium in TBM structure. CB and LL provide for tritium generation. Breeding zone optimum geometry was chosen on the basis of neutronic and thermal hydraulic calculations to provide for: maximum tritium production rate (TPR), minimum MHD pressure drop, maximum CB and LL temperatures with TBM fixed overall dimensions and limitations on materials temperature. As a result the following improvements were achieved in comparison with the basic variant: decrease of MHD pressure drop by 40%, increase of TPR by 4%, increase of temperature rise-up in CB by 24% and in LL by 28%. Test matrix in ITER was estimated for TBM optimum geometry with regard to LL flow rate and inlet temperature variation. Direct and reversed LL flows are considered for ITER tests allowing for a wider range of rear CB layers maximum temperatures: 750-790ºC for direct flow and 800-865 ºC for reverse flow. TBM 3-D thermal hydraulic analysis was performed to reveal some peculiarities of heat exchange between two coolants: LL and helium. Design improvements are suggested to minimize the undesired heat exchange: outlet He collector facing the rear LL duct, heat isolation of inlet and outlet LL collectors.

P1-030 **Development of the tritium breeder monitoring systems for the Lead-Lithium cooled Ceramic Breeder (LLCB) Module of the ITER**

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Demonstration of an electricity production and a tritium breeder is a goal of an ITER mission. A concept of experimental estimation of the tritium-breeding dynamics in the lead-lithium cooled ceramic breeder (LLCB) TBM of the ITER has been developed. The systems based on tritium breeder and neutron flux measurements under ITER plasma D-T experiments for the experimental estimation of the values have been proposed. The lithium carbonate sensors for the tritium breeder and the neutron detectors are applied in the system. Pneumatic and mechanical controls are used to deliver the samples in the containers to the TBM and to extract the containers from the module after a plasma pulse. The channel connects the TBM and an operating zone of the ITER for container move during plasma-operational pauses.

In this study the modifications of the container and sensors for pneumatic control are presented. Neutron calculation of the tritium content in the lithium carbonate under reactor irradiation is performed. The heat distribution in the samples and the materials of the channel under reactor irradiation is discussed. The laboratory facility for investigation of the pneumatic parameters and the container move in the channel is proposed and results of initials tests are discussed.

P1-031 **RF DEMO Helium Cooled Ceramic Breeder Blanket**

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RF DEMO studies are a part of the long-term national program to establish areas for possible magnetic fusion applications and to evaluate safety, environment and waste processing aspects of the applications. Fusion development strategy in the RF considers three major steps: an international experimental thermonuclear reactor (ITER), a demonstration power reactor (DEMO) and a commercial power reactor (CPR). The Helium Cooled Ceramic Breeder (HCCB) blanket is based on the use poloidal Breeder Inside Tube (BIT) concept. Small lithium orthosilicate pebble beds (breeder material) placed inside internal cooling tubes. Pebble-bed/porous beryllium (multiplier material) placed space between module walls and external cooling tubes. Cooling is provided by circulating helium at 8 MPa. The tritium produced in the pebble beds is purged by a separate flow of helium (purge-gas) at 0.1-0.15 MPa. The structural material is chromium ferritic-martensitic steel. RF DEMO design description, principle scheme system of helium cooler energy transformation, and the results of neutronic and thermal analysis for HCCB blanket are presented in this paper.

P1-032 **Neutronic study of an innovative natural uranium-thorium based fusion-fission hybrid energy system**

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An innovative design for a water cooled fusion-fission hybrid reactor (FFHR), aiming at efficiently utilizing natural uranium and thorium resources, is presented. The major objective is to study the feasibility of this concept balanced with multi-purposes, including energy gain, tritium breeding and 233U breeding. In order to improve overall neutron economy of the system, the blanket consists of two main kinds of modules, i.e. the natural uranium module (U-Module) and thorium module (Th-Module), which is alternately arranged in the toroidal and poloidal direction of the blanket. This innovative design is based on a simple intuition of neutron distribution: with the alternate geometrical arrangement, energy multiplication by uranium fission, tritium breeding and 233U breeding are performed separately in different sub-zones in the blanket. The uranium module which has excellent neutron economy under the combined neutron spectrum, plays the dominate role in the energy production, neutron multiplication and tritium breeding. Excess neutrons produced by the uranium modules are then used to drive the thorium module (which has poor neutron economy to breed 233U fuel. Therefore, it creates a new free dimension in the design to realize the blanket multi-purposes: high energy amplifying factor, tritium self-sufficiency and reasonable 233U breeding ratio. The COUPLE code developed by INET of Tsinghua University is used to simulate the neutronic behavior in the blanket. The simulated results show that with the volumetric supporting ratio of thorium module about 0.4, the multi purposes are achievable: intermediate energy multiplication M (larger than 9), TBR (larger than 1.05), at the end of the five years refueling cycle, the 233U enrichment ratio of the thorium modules could reach 1.3%. The neutronic analysis results also show that the preliminary design of this innovative FFHR is of great potential to utilize the bred 233U effectively after the initial fuel load of the first five-year operation.

P1-033 **Synthesis and Fabrication of Lithium Orthosilicate Pebbles by Solid State Reaction Process**

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The fusion of one deuterium and one tritium nuclei will produce one helium nuclei, one neutron and thermal energy and the same is being considered for the first generation fusion reactors. Naturally occurring hydrogen contains 140 ppm deuterium and the technologies to separate it from the compounds of hydrogen are well developed. Whereas, natural hydrogen contains only 7.0 x 10–12 ppm tritium and no technology has yet been developed to separate it from the compounds containing hydrogen, because of its extremely low concentration.

Tritium can be produced by nuclear reaction of Li6 isotope with fast neutrons and the natural lithium contains 7.5 % Li6 isotope. Lithium has low melting point and readily react with oxygen, nitrogen and moisture even at room temperature. That's why lithium containing ceramics viz., lithium titanate and lithium orthosilicate enriched in Li6 isotope are considered for tritium production by nuclear reaction with neutron. Both these materials will be used in the form of spherical pebbles in the size range 0.8-1.2 mm with certain specific properties viz. density, porosity, grain size, mechanical strength etc. Lithium orthosilicate pebbles are usually synthesized and fabricated by the molten spray method using silica and lithium oxide as the reactants at 1450oC. It was found that like lithium titanate (Li2TiO3), lithium orthosilicate (Li4SiO4) can also be synthesized and pebbles can be fabricated by solid state reaction process using silica and lithium carbonate as raw materials. The advantage of this process is that the synthesis can be carried out at 750oC and fabricated pebbles can be sintered at 900oC to achieve the desired properties of the pebbles. Bothe synthesis and sintering temperatures are lower than that of molten spray method. The experimental details and results will be discussed in this paper.

P1-034 **Characteristics of microstructure and tritium release properties of different kinds of beryllium pebbles for application in tritium breeding modules**

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Beryllium pebbles with diameters of 1 mm are considered to be perspective material for the use as neutron multiplier in tritium breeding modules of fusion reactors. Up to now, the main concept of helium-cooled breeding blanket in ITER project foresees the use of 1 mm beryllium pebbles fabricated by company NGK, Japan. It is notable that beryllium pebbles of other types are commercially available at the market. Presented work is dedicated to a study of characteristics of microstructure and parameters of tritium release of beryllium pebbles produced by Bochvar Institute, Russian Federation, and Company Materion, USA.

P1-035 **Design development and analytical assessment of LLCB TBM in Russian Federation during 2012-2013**

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The activity on the design, analysis and R&D for the test blanket module (TBM) with lead-lithium (LL)eutectic coolant and ceramic breeder (CB) is performed in Russian Federation (RF) according to the Technical program of cooperation between the leading research institutes of India (‘Leader’ of LLCB TBM concept) and RF (‘Partner’). During the period of 2012-2013 the joint efforts of RF and Indian specialists are focused on the development and of TBM basic design with optimal set of parameters (in particular for testing both on H-H and H-D operation phases of ITER machine). This article briefly describes the results of TBM design and analysis which have been obtained by RF specialists (‘NIKIET’ and D.V.Efremov institute) in support of LLCB concept (both DEMO blanket and TBM itself). The main directions of this activity in RF institutes are as follows:

- development of TBM design taking into account the ability to manufacture the TBM elements (load-bearing casing, tritium breeding zone, attachment system);
- thermal analysis (both in stationary and transient approach) of TBM design options (four variations of helium and eutectic flowing directions);
- structural analysis of TBM design elements for Inductive I operation mode;
- recommendations (basing upon the results of comparative analysis) on the reference design to be used on further stages of concept development; The critical issues and further plans on the development LLCB TBM and corresponding DEMO blanket in Russian Federation are also presented in this article.

P1-036 **Experimental Investigation of Liquid-Metal Distribution in MHD Flows in Insulating Parallel Ducts**

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The presence of a strong magnetic field can affect a flow distribution of a liquid-metal coolant in a fusion blanket, due to magnetohydrodynamic (MHD) effects. A flow imbalance in the poloidal parallel ducts affects the heat removal performance, which is closely related to the blanket feasibility and safety. The 3D-MHD effects are dominant in such a complicated passage. The present study was performed to experimentally investigate flow distribution characteristics of a liquid metal under the transverse uniform magnetic field with variation of the magnetic field intensity and orientation. The present study employed a test-section of an upward rectangular channel, a U-turn branching area, and two downward parallel rectangular channels with electrically insulating walls. A typical example of the passage geometry can be seen in the DCLL blanket design. The gallium-indium-tin eutectic alloy (GaInSn) was employed as a working fluid. The test-section was made of an acrylic resin. Flow rates in the downward channels were monitored by permanent-magnet electromagnetic flow meters. The Reynolds number was up to 3000. The uniform magnetic field which the electromagnet applied to the manifold that was approximately 1000 mm long, 70 mm wide, and 50 mm deep. The manifold was mounted entirely within the uniform magnetic field. The magnetic field intensity was up to 0.85 T, corresponding to Hartmann number up to 130. The present study shows that as the magnetic field is applied more perpendicularly with respect to the U-turn, much amount of the fluid flows in the inner downward channel than that in the outer one. In addition, 3D numerical simulations were performed to help us to better understand the MHD flow characteristics in the present passage geometry. These results as well as the experimental ones will be presented.

P1-037 **Development of beryllide pebble fabrication as advanced neutron multiplier**

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Advanced neutron multipliers with lower swelling and higher stability at high temperature are desired in pebble-bed blankets, which would have a big impact on the design of a demonstration (DEMO) reactor. Beryllides are one of the most promising candidates as advanced materials. However, few studies on developments of mass production found. It should be clear that the beryllide granulation is available with low cost and high efficiency because of requiring 200-400 tons of neutron multiplier for its breeder blanket. We suggested a new beryllide granulation process, which was combinational process with a plasma sintering method and a rotating electrode method (REM). The plasma sintering is simple, easy to control and can reduce the time of beryllide electrode fabrication by 30% shorter time than the hot isostatic pressing method as conventional methods. The REM has a lot of experiences not only for beryllium granulation in the fusion field but for fabrication of metallic pebble in the general industry. The prototype pebbles were successfully fabricated by the REM using the plasma-sintered beryllide electrode. The beryllide electrode fabrication process was investigated in light of the mass production. From the optimization results, it was revealed that beryllide pebbles with the identical phase composition could be fabricated regardless of the difference of the phase compositions in the beryllide electrodes sintered for different temperature and time. Because this optimized beryllide electrode indicated higher ductility and was sintered at lower temperature for shorter time, this beryllide electrode seems to be more suitable not only to withstand the thermal shock by arc-discharge during granulation but to produce the beryllide pebble on a large scale. Furthermore, the optimization result could lead to expectation of the time reduction and cost saving for pebble mass production because this result can reduce the time of electrode fabrication by 40%.

P1-038 **Neutron cross section evaluation of Chromium and Iron for Fusion Application**

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The evaluations of neutron cross sections for chromium and iron isotopes are presented in fast neutron energy region from several keV up to 150 MeV. Chromium and Iron are important constituents of structural materials of Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM). The existing evaluations such as ENDF/B-VII.1, JENDL-4.0 and JEFF-3.1 show some discrepancies with the measured data available and are mostly limited to the energy range up to 20 MeV. In recent years a reliable data and an extension above 20 MeV in nuclear data evaluation have been required to develop facilities related with fusion energy as well as accelerators. Additionally, there has been an increasing demand from nuclear research, industry, safety and regulatory bodies for nuclear covariance data. This work aims at providing the reliable evaluation data and their covariances through reproducing the differential and integral measurements well. The physical quantities including cross sections and energy-angular distributions were calculated by using the nuclear reaction model code EMPIRE-3.1. The covariance data were generated through the KALMAN code implemented in the EMPIRE system which uses the sensitivity matrices by variations of model parameters and the available measurements. Finally, the benchmark tests were performed by a MCNP5 Monte Carlo code.

P1-039 **Development of CAD-based Discrete Ordinates Code and Comparison of Neutron Flux Distributions in the Korea Helium Cooled Ceramic Reflector Test Blanket Module**

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Korea has been developing a solid-type Test Blanket Module (TBM) called a Korea Helium Cooled Ceramic Reflector (HCCR) TBM to be adopted in the ITER. For neutronics analyses, the MCNP code has been used. MCNP is a well-known Monte Carlo code and has been widely used in an ITER for neutronics analyses since it can handle a complex geometry without any assumptions or simplifications. However, the fluxes and responses are calculated at pre-selected locations (tallies).

Compared to the Monte Carlo method, the discrete ordinates method, referred to as the SN method, has difficulties in modeling a three-dimensional complex geometry. A number of computer codes that use the SN method require a regular mesh (rectangular, cylindrical, or spherical) to model the geometry. Using such a specific regular mesh leads to the simplest difference equations but may require an excessive number of mesh points to adequately model complicated geometries. However, the code under development by KAERI uses an unstructured tetrahedral mesh, and thus it can be applied to solve the radiation transport in a complicated geometry. In addition, the geometry modeling process has become much easier because computational tetrahedral meshes are generated based on the CAD file in the pre-processing stage. As our first phase of applying the code to a TBM neutronics analysis, the neutron flux distribution in the Korea HCCR TBM is compared with that of MCNP, and visualized in a three-dimensional system domain. Visualization of the fluxes and associated reaction rates in the whole system with a single run is one of the merits of a deterministic method and is very useful to check hot spots.

P1-040 **Long-term annealing of lithium orthosilicate based ceramic breeder pebbles**

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Two-phase tritium breeder pebbles, consisting of lithium orthosilicate and lithium metatitanate, were produced by a melt-based fabrication. In order to examine their long-term stability, the pebbles were annealed under reference purge gas atmosphere at 900 °C. The samples were placed in platinum crucibles and purged with 1000 ml/h He+0.1%H2 inside a gas tight aluminum tube at a constant absolute pressure of 1200 mbar. During annealing, the oxygen and water contents were continuously monitored by an optical oxygen sensor and an alumina moisture sensor at the inlet and outlet of the tube. Samples were taken after 4, 32, 64, and 128 days of annealing. The microstructure and the phase content of the annealed pebbles were examined by scanning electron microscopy and X-ray diffractometry, respectively, and were compared to the as-received state. To complement the X-ray diffractometry analysis, the chemical composition of the five samples was checked by X-ray fluorescence analysis and inductively coupled plasma optical emission spectrometry of the main constituents, lithium, silicon and titanium. Also the open and closed porosity were measured by He- and Hg-porosimetry. Additionally, the evolution of the pebbles' surface area was examined by multipoint BET nitrogen adsorption measurements. Crush load tests of single pebbles were used to study possible changes in the mechanical behavior of the pebbles.

P1-041 **Comparison of coating processes for the development of aluminum-based barriers for blanket applications**

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Reduced activation ferritic-martensitic steels (RAFM), e.g. Eurofer 97, are envisaged in future fusion technology as structural material. In the HCLL design for blankets (helium-cooled lead lithium), a corrosive environment of flowing liquid lead-lithium serves as breeder material. Unfortunately, corrosion testing of bare RAFM steels in flowing Pb-15.7Li revealed dramatic dissolution rates of about 400 µm/year at flow velocities of 0.22 m/s at 550°C. Aluminum-based barriers had proven to be a promising way to protect the structural material from corrosion attack and to reduce tritium permeation into the coolant. However, to produce such protective coatings, different processes were and are still under intensive investigation. Coming from coatings produced by hot dipping aluminization (HDA), the development of processes based on electrochemical methods to produce defined aluminum-based scales on RAFM steels became more important during the last years. Two different electrochemical processes to manufacture such scales were in the focus of interest. The first one is based on electrodeposition of aluminum from non-aqueous, volatile and metal-organic electrolytes and is referred to as ECA process. The other one called ECX uses an ionic liquid for electrodeposition and attracted some attention in the last five years in the field of fusion technology. However, electrodeposited aluminum coatings of steel have to be transformed into the desired Al-Fe scales by an adequate heat treatment. However, all three formerly proposed processes, HDA, ECA and ECX exhibit specific advantages and drawbacks, for example in the field of processability, control of coating thicknesses (low activation criteria) and heat treatment behavior. The aim of this article is to compare these different coating processes critically. Thereby some new results for ECX-process were also presented and occurring development needs for the future will be discussed additionally.

P1-042 **Corrosion susceptibility comparison of Eurofer 97 steel in contact two ceramic breeders lithium silicates**

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In the framework of fusion power systems development within the European Union, one of the tritium breeding blanket concepts selected as reference for the future DEMonstration power plant is the Helium Cooled Pebble Lead (HCPL). This concept uses as breeder a lithiated ceramic and Beryllium as neutron multiplier in form of pebble beds. The well known Eurofer'97 (9%CrWVTa) steel is the reference structural material selected for the European blankets. Nowadays, there are many investigations related with corrosion of RAFM steels in liquid breeders (Pb-Li). However, the study of solid breeders is so far nonexistent and only has been presented in the literature considering the compatibility of the lithium ceramics and stainless steels. Although it is expected that the degradation caused by solids is much lower than those due to PbLi, it is not negligible and should be evaluated properly because is strongly related with the tritium release and so, take important safety implications.

The objective of this work is to perform a systematic study of the corrosion process of Eurofer´97 in order to evaluate the compatibility of this material in contact with Li4SiO4 and Li6SiO5ceramic pebbles. The tests were carried out under the typical working conditions, at 550ºC in vacuum atmosphere up to 2880 hours. Different specimens of Eurofer´97 were immersed in the pebble led and studied after testing at programmed times in order to know its corrosion susceptibility as a function of the time. The results show that, although the weight change was negligible during the test, the microstructure obtained by optical and SEM/EDX indicates an evident corrosion attack that could be attributed to exfoliation process.

P1-043 **An Integrated Mesh Translation Scheme for the High-fidelity Coupling of Fusion Neutronics and TH/SM Analyses**

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The design of components such as blanket modules in a fusion device is achieved through an iterative process by performing a series of sequential neutronics, thermal hydraulics (TH) and structural mechanical (SM) calculations. The fusion neutronics calculations performed by Monte Carlo (MC) codes provide the nuclear heating data on regular meshes superimposed on dedicated geometries. However, TH/SM calculations require nuclear heating data to be mapped on structured or unstructured meshes adapted to specific components. Therefore an appropriate scheme for the mesh data transfer has to be implemented for the different calculations. For this purpose, a mesh translation scheme has been developed. In this scheme, superimposed mesh tallies provided by MC codes are automatically processed and tailored into nearly arbitrary structured or unstructured TH/SM meshes. The standard CFD General Notation System (CGNS) format is used for storing the heating mesh data. Available open-source utilities enable the interpolation of the regular MC mesh data into the target TH/SM meshes, using two interpolate schemes: the point-to-point and the volume-weighted scheme. This high-fidelity coupling scheme was implemented as a module and integrated into the open source SALOME computation platform. Therefore mesh translation operations and mesh/results visualization can be performed utilizing the SALOME Graphical User Interface with Visualization Toolkit (VTK) based tools.

The paper presents in details the mesh translation approach and its integration into the SALOME platform. For verification purposes, the Monte Carlo transport calculations were carried out for a simplified blanket model, which was derived from the engineering design of the Helium Cooled Pebble Bed Test Blanket Module (HCPB TBM). The volume integrated heating results obtained from the interpolated meshes were compared with the cell-based integral heating tally results provided by the MCNP calculation. The good agreement achieved for these results demonstrate the accuracy and reliability of the developed mesh translation scheme.

P1-044 **Experimental Investigation of Thermal Properties of the Li_4SiO_4 Pebble Beds**

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Tritium breeder materials, which have the ability to react with neutrons and produce tritium, are required to fuel fusion reactions. Using the lithium ceramic in form of pebble bed is a promising concept for fusion blankets, and worldwide efforts have been devoted to its R&D. Principal features of the Chinese Helium Cooled Ceramic Breeder Test Blanket Module (CN HCCB TBM) concept are the use of the Lithium Ceramic Breeder and the Beryllium Multiplier in form of pebble bed, which are separated and cooled by cooling plates. For the HCCB blanket Lithium Orthosilicate (Li_4SiO_4) is considered as reference material and Lithium Metatitanate ((Li_2TiO_3) as alternative material.

The thermal properties of ceramic pebble beds are important input parameters for the thermo-mechanical design of solid breeder blankets. The objective of this study is to measure the thermal parameters of Li_4SiO_4 pebble beds using the transient plane source method (TPS). The thermal conductivity, thermal diffusivity and specific heat capacity are simultaneously determined from a single measurement process. The thermal parameters of the bed with non-compressed load were measured at temperatures ranging from room temperature to 700 ^{0}C. The packing fraction was about 60% for the single size pebble beds. Helium at atmospheric pressure was used as a filling gas. The effective conductivity was measured as a function of temperature. The experimental results showed that the effective thermal conductivity increased with the increase of the average bed temperature. The measured values showed good agreement with Schulunder's correlation and Hall-Martin's correlation.

P1-045 **Novel Granulation Method for Advanced Tritium Breeder**

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Demonstration power reactors (DEMOs) require advanced tritium breeders that have high stability at high temperatures. Lithium titanate (Li2TiO3) is one of the most promising candidates among tritium breeders. However, a decrease in lithium mass of Li2TiO3 with time occurs in such environments as the DEMO blanket because of Li evaporation and Li burn-up. Therefore, an original material of Li2TiO3 with excess Li (Li2+xTiO3+y) as an advanced tritium breeder that can make up to the lithium loss has been proposed.

In order to fabricate the advanced tritium breeder pebbles, the raw material powder of Li2+xTiO3+y is necessary. As conventional methods, the solid phase reaction using Li2CO3 and TiO2 and the solution phase reaction using an alcohol of Li and Ti have been proposed for Li2TiO3. However, these processes were not suitable for synthesis of Li2+xTiO3+y. As the result of trial and error, the new synthesis process of Li2+xTiO3+y powder has been developed by the solid phase reaction using LiOH(H2O) and H2TiO3. It has been revealed that this advanced material was not easily reduced with hydrogen and has excellent chemical and physical stabilization at high temperatures.

Using this original material of Li2+xTiO3+y, prototype pebble fabrication was applied by an emulsion method as a new granulation process, because it has enough experience in the general industry. From the result of the optimization of granulation conditions, the prototype pebbles of Li2+xTiO3+y with various diameters that has excellent grain size (<5 micrometer) and high sphericity were successfully fabricated. This innovative Li2+xTiO3+y pebbles has contributed for an excellent DEMOs blanket design and for early realization of the DEMOs. This present paper describes novel granulation method of the advanced tritium breeder using these methods including synthesis and granulation techniques.

P1-046 **Effect of plasma sintering consolidation on reactivity of beryllium**

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Beryllium-Titanium intermetallic compound (beryllide) is a promising candidate as advanced neutron multiplier owing to higher stability and lower reactivity at high temperature. It was clarified in the previous studies on synthesis of beryllide that plasma sintering as non-conventional consolidation process could easily and rapidly synthesize the beryllide. In this study, to evaluate the effect of the plasma sintering consolidation of the material on fundamental property, the reactivity with oxygen at high temperature was examined using beryllium metals. The plasma-sintered beryllium (Be-PS) was prepared for comparison of commercial grades of berylliums, that is, S65C and S65E, because it is well-known that the plasma sintering can facilitate to not only eliminate impurities but activate the powder surface due to applying a pulse current. The weight gain and optical surface observation result obviously clarified that the Be-PS and S65E exhibited the higher oxidation resistance than S65C which has larger grain size. Accordingly, it was obvious that the smaller grain size the beryllium has, the better oxidation resistance it has. In addition, the electron probe micro-analysis clearly proved that impurity was apt to be intensively located near grain boundary as an oxide type and the Be-PS contained less impurity than others. Therefore not only grain size but also impurity seems to have a close correlation on reactivity at the high temperature. Since these effects became dominant in the reactivity with water vapor, they indicated similar tendency of H2 generation rate. We report on the results of stream chemical reactivity of different berylliums.

P1-048 **Results of EUROFER-97 corrosion tests in lead-lithium alloy**

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The aim of the work is the experimental evaluation of EUROFER-97 corrosion rate in static and dynamic eutectic lead-lithium (LL) alloy at 550°C with tests duration up to 3000 hrs. Lithium content in the alloy was 15.7% (mass). Rotating disc method was used for dynamic tests with average velocity of specimens 0.5 plus minus 0.15m/s. Specimens were made as 60 degrees sectors of a ring with thickness 3.8 mm. Samples with (and without) protecting coatings were tested. Each disc had two uncoated and 4 coated samples. Three identical sets of the samples were tested at three test sections (RD-1, RD-2, RD-3) under identical temperatures and rotating speeds. LL alloy was placed into molybdenum crucibles. The fourth set of the samples was tested into static LL at the same temperature as in RD. All tests were made under argon atmosphere. Samples in RD-3 test section were tested for 3000 hours, after every thousand hours they were inspected, cleaned from alloy, weighted and measured. Samples in RD-1 were tested for 1900 hours without intermediate inspections. Tests in RD-2 and in static conditions were performed for 3000 hours without intermediate removal as well. Preliminary test results in RD-1 and RD-3 show that maximum corrosion rate takes place during the first thousand hours. For uncoated samples it was around 400 ìm/y, for coated – 180-270 ìm/y. Later on it became approximately constant – 120-130 ìm/y for coated samples. Chemical analyses of LL alloy and its oxide phase, oxygen activometry, optical and electron microscopy, laser spectrometry and microhardness measurements were fulfilled.

P1-049 **Development of the Water Cooled Lithium-Lead Blanket for DEMO**

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Within the EU, several blanket concepts have been considered in the past as possible candidates for DEMOnstration and fusion power plants. In the framework of the Power Plant Conceptual Study (PPCS), five reactor models were considered. Among them, the PPCS model A, featuring a Water Cooled Lithium Lead (WCLL) blanket, is based on near-future technology requiring small extrapolation from present-day knowledge both on physical and technological aspect. In 2010 a new EFDA department Power Plant Physics and Technology (PPPT) was established, having as main objective the development of a DEMO whose conceptual design is scheduled for 2020. R&D and design activities of a WCLL concept with lithium lead breeding loop have been launched. The WCLL blanket could be indeed considered as an alternative to the reference EU helium-cooled concepts for DEMO early phases. This paper presents a WCLL blanket design adapted to 2012 EU DEMO specifications.

Relatively small modules with straight surfaces are attached to a common rear banana shaped supporting structure housing feeding pipes. Each module features reduced activation ferritic-martensitic steel as structural material, liquid Lithium-Lead as breeder and neutron multiplier and carrier. Water at typical Pressurised Water Reactors (PWR) conditions is chosen as coolant. A preliminary design of the equatorial outboard module has been achieved. Finite Elements analyses have been carried out in order to assess the module thermal and mechanical behavior. Two First Wall (FW) concepts have been proposed one favoring the manufacturability, the other favoring the thermal efficiency. The Breeding Zone has been designed in order to minimize the number of internal cooling pipes. C-shaped Double Walled Tubes have been assumed in order to minimize the Water/LiPb interaction likelihood. The priorities for further development of the WCLL blanket concept are eventually identified in the paper.

P1-050 **Liquid metal magnetohydrodynamic flows in manifolds of dual coolant lead lithium blankets**

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In the framework of the European study of a DEMO fusion reactor a dual coolant lead lithium blanket (DCLL) design is under development. It is based on a helium cooled ferritic steel structure, a self-cooled lead lithium breeding zone and the use of low-conducting channel inserts for thermal and electrical insulation. The inserts serve to electrically decouple the liquid metal flow from the steel wall in order to reduce the total induced current density and associated magnetohydrodynamic (MHD) pressure drop.

In the past several studies have been performed to investigate liquid metal flows in long poloidal channels, as considered in DCLL blanket concepts. It has been shown that with efficient electrical insulation the MHD pressure drop in such ducts is not an issue. However, the major fraction of pressure losses arises in 3D geometric elements that distribute the liquid metal into the larger breeding zone, like e.g. expansions and manifolds. Here due to the occurrence of complex electric current loops MHD interactions are particularly intense and electromagnetic forces become dominant. This may result in non-uniform flow distribution in the parallel vertical channels that are fed by the manifold. Application of insulation at manifold walls, as used in long ducts, is ineffective, since currents close their path inside the fluid cores. In the present paper liquid metal MHD flows are studied for different design options of a DCLL manifold with the aim of optimizing the flow partitioning in poloidal ducts and reducing MHD pressure losses to acceptable values. Moreover, the occurrence of flow detachment and closed recirculations is analysed, since their presence could be detrimental to adequate tritium removal.

P1-051 **Development of the Helium Cooled Lithium Lead Blanket for DEMO**

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The Helium Cooled Lithium Lead (HCLL) blanket is one of the candidate European blanket concepts selected for the DEMOnstration fusion power plant that should follow ITER. In a fusion power plant, the blanket is one of the key components since it has to withstand extremely severe operating conditions while insuring tritium self-sufficiency, adequate neutron shielding and coolant temperatures suitable for en efficient power conversion cycle. Its design has therefore a significant impact on the plant performance, availability, safety and economics. In 2012, the European Fusion Development Agreement (EFDA) agency issued new specifications for DEMO, mainly consisting in updated plasma energy source parameters. This paper describes the work performed to adapt the previous 2007 HCLL-DEMO blanket design to the new specifications and the continuing work of improving the blanket concept. A new segmentation for the inboard and outboard blanket has been defined assuming straight surfaces for all blanket modules. Blanket segmentation follows the previous Multi Module Segment (MMS) option, where modules are attached to a common back supporting structure which also serves as manifold for Helium and PbLi distribution. A detailed CAD design of the central outboard module has been defined. Thermo-hydraulic and thermo-mechanical analyses on of the First Wall have been carried out in order to confirm the viability of the new design. Preliminary pressure drop calculations have been performed to assess the required Helium pumping power. For the attachment of the modules to the common backplate, a new solution based on the use of Tie Rods, derived from the design of the corresponding HCLL Test Blanket Module for ITER, has been proposed. This paper also identifies the priorities for further development of the HCLL blanket concept.

P1-052 **Influence of surface oxidation on electric potential measurements in MHD liquid metal flows**

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Study of liquid metal flows in strong magnetic fields plays an essential role in the development of fusion reactors where breeding of tritium and partial heat extraction are accomplished by lead lithium circulating in the blanket. Laboratory experiments have been performed to investigate liquid metal flows in rectangular channels under the influence of intense magnetic fields. In the test section walls parallel to the field are electrically insulating and those perpendicular are covered by a thin copper layer to achieve a well-defined electric conductance. Electric potential has been recorded at different locations on the surface of the duct. First measurements showed a perfect agreement with numerical and analytical solutions.

A second series of experiments has been carried out after opening the test section due to required technical modifications. Subsequent measurements showed a modified potential distribution compared to previous data. Discrepancies have been ascribed to the occurrence of contact resistance by oxidation of the copper surface during opening of the loop. The influence of a wall oxide film on electric potential measurements has been described by an analytical solution and by numerical simulations. The possibility of accounting for the influence of a contact resistance allows giving a proper interpretation of the measured electric potential distributions.

P1-053 **Influence of magnetic field deformation by ferromagnetic wall materials on MHD flows in pipes and ducts of fusion blankets**

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Several liquid metal blanket concepts are considered for applications in future fusion power reactors. Depending on the design options, the liquid metal can serve primarily as breeder material or it can play additionally the role of a coolant. For a reliable blanket design the magnetohydrodynamic (MHD) interaction of the liquid metal with the strong magnetic field confining the fusion plasma has to be fully understood.

Current designs of liquid metal blankets like e.g. the European helium cooled lead lithium blanket (HCLL), foreseen as a test blanket in ITER, rely on EUROFER as wall material. Due to its ferromagnetic properties the magnetic field inside pipes and ducts will be modified in comparison with the applied one with possible consequences on the liquid metal flow. In general, wall materials with higher magnetic permeability concentrate magnetic field lines in the walls and shield, but only partly, the interior of the duct. Initially straight field lines may become curved or strongly deformed.

In order to describe such phenomena a 3D numerical code based on finite volume technique is under development at the Karlsruhe Institute of Technology. The electrodynamics problem is solved in terms of a magnetic vector potential and electric potential formulation of the governing equations. First results are presented as validation of the implementation for the case of fully developed MHD flows in a ferromagnetic pipe, for which an analytical solution for the modified magnetic field and the resulting changes of the MHD flow is available in literature. Encouraged by the good comparison between the analytical solution and numerical results in the test case, the code is applied to determine the flow in geometries for which analytical solutions are not available.

P1-054 **Status of the new DEMO HCPB Blanket design in the European DEMO studies**

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In the frame of the activities of the European DEMO Breeder Blanket Programme, the Karlsruhe Institute of Technology (KIT) is supporting the design and qualification of the Helium Cooled Pebble Bed Blanket Module (HCPB-BM). For the new reactor study started under the EFDA PPP&T department, the existing concept of the HCPB blanket necessitates to be adapted to the new DEMO configuration and geometry. In the frame of these activities, previous studies have been revised and a new set of boundary conditions has been elaborated to guide the new conceptual design. This paper presents the new CAD design developed in this study and the results of the thermo hydraulic and thermo mechanical investigations performed for its validation. The CAD design presents a proposal of segmentation for an inboard and an outboard blanket segment arranged according to the Vertical Maintenance System. The design of the blanket for a reference equatorial box have been developed including First Wall, Breeding Zone, internal Stiffening Grid and Manifold system. All of these components have to withstand different typologies of loads, as thermal and pressure loads, during both normal and off-normal operation, e.g. LOCA (Loss of Coolant Accident). The general helium cooling system inside the segments is described and assessed. Heating and coolant parameters have been defined and these parameters have been used to develop a 3D thermal hydraulic model and perform corresponding computations with the code STAR CD®. The aim of the performed analyses was to evaluate the heat transfer conditions inside the First Wall, and hence the temperature behavior along the channel. The outcomes of the thermal hydraulic calculations have been considered as input boundary conditions in the ANSYS WB thermo- structural model to calculate several loading conditions under primary and secondary boundary conditions. The paper introduces a parametric dimensioning study of central manifold behind the blanket system. The assessed results are presented according to the SDC IC and RCC-MRx codes.

P1-055 **Tritium permeation experiments using reduced activation ferritic/martensitic steel tube and erbium oxide coating**

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Controlling tritium in blanket systems is one of critical issues in DEMO and commercial fusion reactors from the viewpoint of fuel cycle and radiological safety. In order to suppress tritium permeation through structural materials, ceramic coatings have attracted increased attention as a tritium permeation barrier (TPB). Hydrogen isotope permeation through coated samples has been experimented for several decades mostly with hydrogen or deuterium at a high partial pressure in order to detect a small permeation flux. However, the number of literatures in which tritium permeation has been measured on the blanket condition is limited. In the present study, tritium permeation experiments using reduced activation ferritic/martensitic (RAFM) steel tubes with and without erbium oxide (Er2O3) coating as a promising candidate for the TPB were carried out for understanding the tritium permeation behavior at low tritium partial pressure in the framework of the Japan-US joint research project TITAN (Task 1-2). Er2O3 coatings were prepared by dip-coating technique on RAFM steel F82H tubes. The fabrication procedure was repeated twice for a better coverage of the coating, and the thickness was approximately 0.2 micron. Tritium permeation experiments were performed with 1–100 ppm (0.1–10 Pa) tritium and helium mixture by a gas-driven permeation setup at Idaho National Laboratory. Tritium permeability of the uncoated sample agreed with hydrogen/deuterium results and indicated the surface effect at low tritium partial pressure from the pressure dependence. The coated sample showed the tritium permeation only at more than 873 K because of its long diffusion time. The permeability showed three orders of magnitude lower than that of the uncoated substrate at up to 973 K only with the 0.2 micron-thick coating. In addition, the memory effect of the ionization chamber was observed during the experiments, suggesting that tritium permeated through the coating via HTO.

P1-056 **Microstructural Characterization for Radiation Enhanced Deuterium Loaded RB-SiC.**

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In liquid metal (LM) blankets, flow channel inserts (FCI) will have to be introduced in order to assure thermal and electrical insulation between the steel and the LM, and to minimize magneto-hydrodynamic (MHD) pressure loss. In addition to low thermal and electrical conductivity, the FCI materials must also have low tritium absorption in order to guarantee tritium self-sufficiency and also to keep the tritium inventory below the requested limits. SiC-derived materials are primary candidates for FCI due to their excellent properties. During reactor operation the ceramic material will be exposed to tritium and also to a hostile radiation environment. Absorption, diffusion, and desorption of tritium is expected to occur and these processes will strongly depend on the irradiation conditions, neutron flux and purely ionizing radiation. In particular radiation enhanced absorption and reaction between tritium and the ceramic material is expected to occur due to the ionizing radiation component.

The main aim of the work to be presented is to characterize the ionizing radiation induced absorption of hydrogen isotopes in SiC and to identify the hydrogen related centres formed during the irradiation process. Reaction bonded (RB) SiC samples were deuterium loaded for different times and pressures. In addition deuterium loading was carried out with both the sample and the surrounding deuterium gas exposed to 1.8 MeV electron irradiation in order to evaluate the radiation enhanced deuterium absorption. Both electron irradiated and unirradiated samples were analyzed by SEM and ATR-IR techniques in order to study the changes occurred on the surface as microstructural or Si-C bonding modifications due to experimental conditions. Radiation enhanced absorption and reaction between deuterium and SiC have been addressed and characterized. The results indicate that the radiation enhanced tritium absorption by SiC flow channel inserts could seriously affect tritium self-sufficiency and inventory in future fusion reactors.

P1-057 **Conceptual design of a water cooled breeder blanket for CFETR**

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China Fusion Engineering Test Reactor (CFETR) is an ITER-like superconducting tokamak reactor whose conceptual design is being conducted. Its major radius is 5.7m, minor radius is 1.6m and elongation ratio is 1.8. Its mission is to achieve 50 -200 MW of fusion power, 30% - 50% of duty time factor, and tritium breeding to ensure the self-sufficiency. As one of breeding blanket candidates for CFETR, a water cooled breeder blanket with superheated steam is proposed and its conceptual design is carried out. In this design, sub-cooling water of 7MPa (265 ^{o}C) is fed into cooling plate in breeding zone and is heated up to boiling steam of 285 ^{o}C, and then is superheated up to 450 - 500 ^{o}C in First Wall (FW). Due to low density of superheated steam, it has negligible impact on neutron absorption in FW so that the high energy neutron entering into breeder zone may be moderated by water in cooling plate help enhancing tritium breeding by 6Li(n,α)T reaction. Ceramic breeder (Li2TiO3) pebble and the neutron multiplier Beryllium metallic compounds (Be12Ti) pebble are chosen, because Li2TiO3 and Be12Ti are expected to have better chemical stability and compatibility with water in high temperature. However, Be12Ti may lead to a reduction in tritium breeding radio (TBR). Furthermore, a spot of sintered Be plate is used to improve neutron multiplying capacity in a multi-layer structure. As one alternative option, in spite of less TBR, Pb is used to replace Be plate in viewpoint of safety. In this contribution, study on neutronics and thermal design for a water cooled breeder blanket with superheated steam is reported.

P1-058 **Corrosion and Transport of Activated Corrosion Products in DCLL Blanket**

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In the Dual-Coolant Lead-Lithium (DCLL) blanket, lead-lithium alloy (PbLi) is used as a breeder and coolant while reduced activation ferritic/martencitic (RAFM) steel is used as a structural material. A flow channel insert (FCI) made of silicon carbide (SiC) is utilized as electrical and thermal insulator to reduce the magnetohydrodynamic (MHD) pressure drop and to decouple hot PbLi from the RAFM structure. The flow-induced corrosion of RAFM walls occurs in the thin 2-mm gaps between the FCI and the outer RAFM duct and is strongly dependent on the velocity and the temperature field, which in turn are affected by a plasma-confining magnetic field. The corrosion products, including activated materials, are then transported with the flowing PbLi from the blanket into the ancillary system (e.g. a heat exchanger) where they can precipitate. It turns out that precipitation of radioactive isotopes, such as Cr-51, Fe-59, Co-58, Mn-54 and especially Co-60, is of special concern since contamination due to activated corrosion products (ACP) may lead to difficult maintenance issues during shut-downs and can significantly increase the associated maintenance costs due to increased individual and collective doses to meet personnel and increased shielding requirements. These coupled fluid flow, MHD, heat & mass transfer and activation processes in the DCLL blanket are analyzed simultaneously to address both the corrosion losses and ACP transport.

The newly developed TRANSMAG (TRANSport phenomena in MAGnetohydrodynamic flows) code is used for MHD and heat & mass transfer calculations, while the neutron induced radioactivity in ACP is computed by means of the EASY (European Activation SYstem) code, using ad-hoc neutronic calculations. The main features of the TRANSMAG code are described, with special attention to the aspects of corrosion losses and ACP transport and deposition. Obtained computational results for corrosion, activation and transport of activated corrosion transport under DCLL DEMO blanket conditions are discussed.

P1-059 **Fission blanket benchmark experiment on spherical assembly of uranium and PE with PE reflector**

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New concept of fusion-fission hybrid for energy generation have been proposed. To validate the nuclear performance of fission blanket of hybrid, two types of fission blanket assembly was setup and measurement was made of the reaction rate distribution for uranium in the spherical assembly of depleted uranium and polyethylene by Plate Micro Fission Chamber(PMFC). There are two PMFCs in experiment, one is depleted uranium chamber and the other is enriched uranium chamber. The material in the depleted uranium chamber is strictly the same as the material in the assembly. In this work, The Monte-Carlo transport code MCNP5 and continuous energy cross sections ENDF/BV.0 were used for the analysis of fission rate distribution in the two types of assemblies. The calculated results were compared with the experimental ones. The overestimation of fission rate for DU and EU were found in the inner baundry of the two assemblies. However, the C/E values tends to decrease for the distance from the core slightly and the results for EU is better that of DU.

P1-060 **Measurement and Calculation of Neutron Energy Spectrum in an Alternate Depleted Uranium-Polyethylene System**

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In order to check calculation methods and programs for the concept design of subcritical blanket in hybrid fusion-fission reactor, the neutron energy spectra were measured on alternate depleted uranium/polyethylene shells with 14 MeV neutrons using the recoil proton technique with a BC501 scintillation detector. The standard experimental uncertainties of neutron energy spectra were 6.0%~7.9% in the region of 1MeV~14 MeV and 5.6%~7.5% in the region of 2.5MeV~16MeV. The calculated results with the MCNP/4B code were compared with the experimental results. It showed that the calculated results were agreed with the experimental results within the range of uncertainties at the majority of measurement points.

Topic C Fuel Cycle

P1-061 **Theoretical prediction of thermodynamic properties of tritiated beryllium molecules and application to ITER source term**

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To evaluate the safety of any nuclear installation, one has to predict for any possible accidental situation the amount and the chemical speciation of all the chemically toxic and/or radio-toxic compounds (often called source terms) that can be released to the environment. We consider two reference accidents in the ITER fusion installation, the air and water ingress in the vacuum vessel. For these accidents the radiologic hazard, due mainly to tritium, is aggravated by the chemical toxicity of beryllium that will be mobilized. Discharge lines of the vacuum vessel are equipped with filtering devices. To assess their efficiency and to evaluate the consequence on the environment if these devices failed to trap these species, one has to be able to determine the evolution of the chemical speciation of all tritiated and/or beryllium compounds, during their transport from the vacuum vessel to the environment. As a first step, this evolution can be computed from the thermodynamic equilibrium of the Be-W-O-H-T system. We used quantum chemistry to fill the lacks or to assess the thermodynamical properties of the beryllium species, avoiding classical experimental determination that would have been made complex and costly by the toxicity of beryllium. We proved by a comparison with results obtained from highly reliable post Hartree-Fock methods that our methodology based on density functional theory allows to appropriately describe properties of beryllium molecules in the gas phase. Then, our methodology has been used to calculate structural and thermo-chemical properties of all species of the Be-W-O-H-T system that can be formed in these accidental conditions. These results will complete the thermo-chemical database of the IRSN code, ASTEC, devoted to the simulation of the progress and the evaluation of the consequences of accidents in nuclear installations.

P1-062 **Investigation on degradation mechanism of ion exchange membrane immersed into high-concentration tritiated water under the Broader Approach Activities**

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The ion exchange membrane such as Nafion® is a key material for electrolysis cells of the Water Detritiation System. Endurance of ion exchange membrane immersed into high-concentration tritiated water has been demonstrated under the Broader Approach activities, as a R&D on endurance of fuel cycle components at high tritium exposure. Long-term exposure of Nafion® ion exchange membrane into 1.38x1012Bq/kg of tritiated water was conducted at room temperature for up to 2 years. The ionic conductivity of Nafion® ion exchange membrane after immersed in tritiated water was changed. The change in color of membrane from colorless to yellowish was caused by active radical reactions. Infrared Fourier transform spectrum of the membrane immersed in tritiated water revealed a small peak for bending vibration of C-H situated at 1437cm-1 demonstrating the formation of hydrophobic functional group in the membrane. We have previously investigated the ion exchange capacity of membrane immersed in tritiated water was not decreased. In addition, the peak for bending vibration was clearly eliminated in the spectrum of the membrane after treatment with an acid. The high-resolution solid state 19F NMR spectrum of the membrane after immersed in tritiated water was similar to that of membrane irradiated with gamma-rays. From the 19F NMR spectrum, any distinctive degradation in the membrane structure by interaction with tritium was not measured. It was interpreted as the decrease in ionic conductivity was caused by free hydrophobic products that were trapped in the membrane.

P1-063 **Hydrogen and water vapor adsorption properties on cation-exchanged mordenite for use to a tritium recovery system**

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Tritium recovery system using adsorption, cryosorption or isotope exchange reaction has already been proposed for a solid breeding blanket system of a nuclear fusion reactor. Synthetic zeolite is often used as an adsorbent for water vapor recovery and for hydrogen isotopes recovery at low temperature, and is also applied as a carrier of chemical exchange catalyst for hydrogen isotopes recovery. And, it is well known that the properties of synthetic zeolite are changed easily by exchanging its cation.

Synthetic mordenite used in this work as the start material is one of synthetic zeolite having large silica/alumina ratio, and has Na ion as exchangeable cation (Na-MOR). Many investigators including the present authors have investigated hydrogen adsorption properties of cation-exchanged mordenite with alkali metal ions or alkali earth ions so far. And, the present authors also have reported that cation-exchanged mordenite with Ca ion (Ca-MOR) indicated fairly large hydrogen adsorption capacity at 77 K in comparison with Molecular Sieves 5A (MS5A). So, in this work, hydrogen adsorption properties of cation-exchanged mordenite with transition metal ion were investigated mainly. The cation-exchanged mordenite with Ag ion (Ag-MOR) has indicated significantly large hydrogen adsorption capacity in lower pressure range at 77 K in comparison with Ca-MOR. The discussion from the viewpoint of adsorption rate is still remaining, however, more compact cryosorption bed for tritium recovery system is possible to design if Ag-MOR is adopted.

Thermal desorption properties of water vapor from cation-exchanged mordenite also have been investigated. It was shown that chemisorbed water on the cation-exchanged mordenite with Li ion (Li-MOR) significantly increased in comparison with the start material.

P1-064 **On ion implantation and damage effect in $Li2TiO3$ as a fusion breeder blanket: a technological approach for in-situ degradation testing.**

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After being generated by nuclear reactions in the candidate ceramic breeder blanket (BB), Tritium diffuses reaching the material surface where it desorbs, in combination with any O or H present. Because of the experimental difficulty working with it, other alternatives to study the gas mobility and the surface phenomena are been investigated. Actually molecular dynamics simulations and experimental studies have proved [1,2] that Li-ion conductivity is in close relationship with Tritium diffusion. On the other hand light ions, such as the first hydrogen isotope (D) or an element with the same atomic mass and spin (3He), are been chosen to simulate T behaviour.

The present work is the first technological approach to establish a testing in-situ technique able to characterize the degradation of Li2TiO3 as breeder material during operation.

Firstly, in an attempt to find the outgassing/release temperature and to study its bulk diffusion, NRA was used for profiling the 3He and D depth distributions of as-implanted and damaged (by γ-ray and Ti-implantation) samples as a function of the annealing temperature.

Secondly, Li+ mobility was characterized by Impedance Spectroscopy and Surface Conductivity, varying the temperature of the same batch of samples. Considering the relevance of the microstructure (porosity and grain size), the study was supported by SEM observations and Hg-porosimetry measurements.

Concerning D-implanted samples, the results suggest the complete release at temperatures higher than 200ºC. On the other hand, the implanted 3He desorbs continuously with temperature, a 7-8% still being measured at 900ºC. The concentration of trapped 3He is stabilized after 1 hour of annealing at 400ºC.

The conductivity measurements confirm the relevance of conduction paths along grain boundaries as of the OHC (oxygen-associated hole centres) formed by γ -irradiation. Local damage and consequent degradation of the physical properties are caused by low energy particles, depositing their energy very near the surface and producing high levels of ionization.

[1] G.Vitins et al.; J.Solid State Electr. 6 (2002) 311-319

[2] A.Buljian, L.Padilla-Campos; ACI 1(2) (2010) 37-46

P1-065 **Experimental testing results to demonstrate tritium extraction in LiPb loop systems with a compact permeator against vacuum**

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Tritium recovery in fusion reactors is one of the main goals in R&D, as a limited inventory is available and its production is uneconomic. That is the reason why an efficient system is indispensable to be developed to allow this self-sufficiency by means of a compact permeator against vacuum. This permeator has been designed and manufactured, as well as all the necessary equipment for the LiPb loop in order to test and demonstrate that an in-pipe integrated solution is possible and to validate the manufacturing process. Efficient rates for a more optimized future model could be then extrapolated. The aim of this paper is to show the different testing results that have been carried out in this research project. These results include permeation properties of the material considered for the permeator, as long as it has been manufactured with a novel technique, Selective Laser Melting. They also include vacuum tests on the permeator to quantify possible leakages and to set up and analyze the capability to generate vacuum inside the permeator, as well as permeation testing inside the loop considering a gas mixture of hydrogen and argon in the circuit and the last step that considers PbLi circulation in the loop.

P1-066 **Tritium retention properties of tungsten, co-deposited carbon films and graphite**

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For prediction of tritium inventory in future fusion devices, it is important to investigate the tritium retention properties in plasma-facing materials. Tungsten and carbon are primary plasma-facing materials in fusion devices. Carbon materials are easily eroded by incident hydrogen particles and the sputtered carbon atoms are redeposited in vacuum vessel. From the view point of long-term employment of plasma-facing material in a DEMO reactor, the time evolution of amount of retained tritium is needed to be evaluated for a precise estimation of the tritium inventory. In the present study, tungsten, graphite and co-deposited carbon films were irradiated by tritium ions, and then the amount of retained tritium and its reduction with time were investigated.

An ion beam of a mixture of D2+ and DT+ ions with an incident energy of 1 keV were irradiated to the samples at a fluence of 9 x 1016 (D+T)/cm2 (4.5 x 1014 T/cm2). After the irradiation, the samples were preserved in a vacuum and the amount of retained tritium was measured periodically with with γ-ray-induced X-ray spectrometry and imaging plate measurements. The amounts of retained tritium in tungsten and co-deposited carbon films were roughly one half and one seventh of that for graphite, respectively. The amount of retained tritium in graphite did not change during one month period. On the other hand, the decreases of the retained tritium were observed in tungsten and co-deposited carbon films. For tungsten, the amount of retained tritium decreased to approximately half of the initial amount 15 days after tritium irradiation. For co-deposited carbon films, a 15-20 % reduction was observed 40 days after the irradiation. The results indicate that the reduction of the amount of retained tritium must be taken into account for long-term tritium retention in future fusion devices.

P1-067 **Hydrogen Isotopes behavior on water-metal boundary with simultaneous transferring from and to the metal surface**

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Tritium confinement is the most important safety issue in the fusion reactor. Specially, tritium behavior on the water metal boundary is very important to design tritium plant with breading blanket system using cooling water.In order to discuss the tritium behavior, a series oftritium permeation experiment into pressurized water jacket with He or jacket purging <1000ppm of water vapor in Arhave been performedthrough pure iron piping with/without 7 micro-meter gold plating, which contained about 1 kPa of pure tritium gas at 423 K, with monitoring the chemical forms of tritium permeated into water or water vapor jackets[1]Also, deuteriumpermeation experiments from heavy water vessel throughvarious metal piping, such as pure iron (Fe), nickel (Ni), stainless steel (SS304), and pure iron with 10 micro-meter gold plating, wereperformed at573K and at 15 MPa.[2] During all the above experiments, the surfaces of metal piping except gold plating one were oxidized at the water metal boundary and then hydrogen isotopes would generate by the oxidation reactions, such as Schikorr reaction: 3 $Fe(OH)_2$--> Fe_3O_4 + $2H_2O$ + H_2. Recently, using the above heavy water system, we have succeeded to detect simultaneous hydrogen isotopes transfer from and to the metal surface by introducing H2 gas to the metal piping after stabilized deuterium permeation was detected.

This paper summarizes the consideration of the hydrogen isotope behavior on the water metal boundary, which hydrogen isotopes moves from the metal side to water and from water side to the metal, using all of the previous and recent data.

[1] T. Hayashi et al., Hydrogen isotope behavior transferring through water metal boundary,*Fusion Sci. and Technol.,***60** (2011) 369-372

[2] T. Hayashi et al., Hydrogen isotope permeation from cooling water through various metal piping, *Fusion Eng. and Des.*, **87** (2012) 1333-1337

P1-068 **Recent results on tritium technology for DEMO reactor in JAEA under BA program**

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The following R&Ds have been proposed and carried out as important and generic research subjects under BA program. Cryosorption is often employed to gas chromatograph (GC) of hydrogen isotope analysis. We have studied cation-exchanged MORs. It has been shown that MOR with Ca2+ as exchanged cation has fairly desirable characteristics for the material of GC.

To understand plasma surface interactions, tungsten samples were exposed to Low-energy (38 eV/D), high flux (1022 D+/m2/s) deuterium (D) plasma to a fluence of 1026 D/m2. After the plasma exposure, tritium was introduced into the samples by exposure to deuterium-tritium gas mixture. The results of the imaging plate revealed that tritium was concentrated mainly within the area exposed to the D plasma. Autoradiography of the W surface showed that tritium was concentrated on the grain boundary and blisters. The effects of tritium on self-passivation behavior of SUS304 stainless steel in 1 N H2SO4 solution were studied. It was found that there would be three equilibrium: The first equilibrium would be dissolution of Fe, formation of magnetite and formation of H2. After the first passivation, the equilibrium would shift to the second one that would be dominated by formation of hematite from magnetite. The equilibrium then shift to third one that would be dominated by formation of Fe2+ from Fe3+. The ion exchange membrane such as Nafion® is a key material for electrolysis cells of the Water Detritiation System. Long-term exposure of Nafion® ion exchange membrane into 1.38x1012Bq/kg of tritiated water was conducted at room temperature for up to 2 years. The high-resolution solid state 19F NMR spectrum of the membrane after immersed in tritiated water was similar to that of membrane irradiated with gamma-rays. From the 19F NMR spectrum, any distinctive degradation in the membrane structure by interaction with tritium was not measured.

P1-069 **Construction and commissioning of a Hydrogen Cryogenic Distillation system for tritium recovery at ICIT Rm. Valcea**

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Cryogenic distillation (CD) of hydrogen in combination with Liquid Phase Catalytic Exchange (LPCE) or Combined Electrolytic Catalytic Exchange (CECE) process is used for tritium removal/recovery from tritiated water.

Tritiated water is being obtained after long time operation of CANDU reactors, or in case of ITER mainly by the Detritiation System (DS). The Cryogenic Distillation System (CDS) used to remove/recover tritium from a hydrogen stream consists of a cascade of cryogenic distillation columns and a refrigeration unit which provides the cooling capacity for the condensers of CD columns. The columns, together with the condensers and the process heat-exchangers are accommodated in a vacuum insulated coldbox. In the particulary case of the ICIT Plant, the cryogenic distillation cascade consists of four columns with diameters between 100 to 7 mm and it has been designed to process up to 7 mc per hour of tritiated deuterium. This paper will present the steps undertaken for construction and commissioning of a pilot plant for tritium removal/recovery by cryogenic distillation of hydrogen. The paper will show besides preliminary data obtained during commissioning, also general caractheristics of the plant and of its equipments.

P1-070 **Tokamak exhaust gas composition measurement via different mass spectrometers**

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The composition of exhaust gas of a tokamak gives important information concerning the burn up fraction at plasma burn as well as the impurities from wall outgassing or radiative cooling.

For analysing the tokamak exhaust gas, mass spectrometers could be used. But as the plasma pulses can be short (e.g. about 400 s for ITER), fast responding devices are needed, so that several mass scans can be performed during a pulse.

This can e.g. be done by an autoresonant ion trap mass spectrometer (ART-MS), which can determine partial pressures from 1-300 amu in 120 ms. Like this online measurements are possible and also fast pressure changes can be traced. Unfortunately a disadvantage of this device and the fast scan time is a lack of resolution and sensitivity. Therefore a second device, a high resolution quadrupole mass spectrometer, is used for accurately determining the partial pressures of the different exhaust gases. Even helium and deuterium which both appear at a mass of about 4 amu can be separated and quantified. As this of course takes some more time, it can only be performed a few times during a short pulse.

When used in a tokamak environment both devices need to be shielded from magnetic fields.

To check the applicability of these two devices for online-mass spectrometry, measurements of expected exhaust gases are performed at Karlsruhe Institute of Technology (KIT).

P1-071 **Activity monitoring of ppm concentrations of tritium in helium gas streams by beta induced X-ray spectrometry**

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Safe and efficient tritium processing at tritium handling facilities like fusion reactors needs accurate and reliable tritium monitoring systems. Especially for high tritium throughputs like in ITER and DEMO, real-time and inline monitoring tools are desirable. This holds in particular for the activity monitoring in the blanket tritium extraction system, which needs to be sensitive on tritium with ppm concentration in about one bar of helium, thereby insuring measurement uncertainties below five percent.

Beta induced X-ray spectrometry (BIXS) seems to be an applicable method for activity monitoring of gaseous, solid and liquid tritium sources. The present work gives an overview of the physical processes relevant for a BIXS system and describes the development of a BIXS process activity monitoring tool which is optimized to measure ppm concentrations of tritium in helium gas streams of about one bar. Monte-Carlo simulation results are presented regarding the geometry of a sample cell, optimized bremsstrahlung production and the projected sensitivity of such a system. At the Tritium Laboratory Karlsruhe a prototype BIXS-chamber was built according to these simulation results and first tritium measurements were performed. Experimental results on the BIXS-chambers overall detection efficiency and on the memory effect are presented and calibration curves with tritiated test gas mixtures are given. Finally, the applicability of the BIXS method for a blanket tritium extraction system is discussed.

P1-072 **Status & Progress of the R&D Work for ITER Magnet Supports**

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Magnet supports is designed as class A quality components. First of all, selection the suitable technical road for the finial high quality components is important. It is estimated that several thousand tons of various type 316LN stainless steel including hot-rolled plate, forged blocks and pipes, is needed for all the magnet supports. Last year, the large-size forged block for PFS6(total weight up to 10T) was developed domestically. Both the plate and the forged block showed high strength and good elongation at room temperature and 4.2K. In addition, almost no ferritic phase could be seen. A new design for GS manufacture without welding was put out, the static stress analysis using three-dimensional finite element model (FEM), was developed to analyze the redesigned structure. It can be known that the stress in the present load condition/combination is under the stress limitation of the material. In order to further check the engineering stability of this support under various possible work condition, a special test platform, which can simulate all forces during ITER operation, to check the prototype TF pedestal was designed and constructed. Various cooling pipe is needed for maintaining the low temperature of the magnet supports. We develop laser brazing method to reduce the heat input and then improve the connection, and the microstructure observation shows that almost no microcrack could be found. In the ITER magnet system, more than 10000 various bolts and tie rods will be used, the material include Inconel 718, A286 and 316LN. We have successfully developed the fastener fabrication technology. The fastener qualification is an important work to guarantee the structure safety of magnet supports. The engineering test of the fastener, for instance, tensile strength, fatigue at 77K is developed in our institute. The further test is in schedule.

P1-073 **Corrosion of 9Cr-1Mo steel in Pb-17Li in a rotating disc experiment**

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India has proposed Lead–Lithium cooled Ceramic Breeder (LLCB) as the blanket concept for its Test Blanket Module (TBM) to be tested at the International Thermonuclear Experimental Reactor (ITER). A serious problem arising from the use of Pb-17Li as a coolant is the corrosion of containment material and its deleterious effect on the mechanical properties. In general, ferritic –martensitic (FM) steels are expected to serve better under fusion reactor environment since they have lower swelling, develop lesser thermal stresses and lower induced activation as compared to the austenitic varieties. Nevertheless, to establish a particular FM steel as a suitable structural material, corrosion compatibility needs to be studied thoroughly under various conditions of temperature and flow velocity. For analyzing the corrosion behavior of various potential structural materials in Pb-17Li, a magnetic coupling based rotating disc test facility has been developed. The set up provides scope for preliminary screening of materials with better compatibility and allows studying the effect of flow velocity on the corrosion behavior of Pb-17Li at a given temperature. The present work describes the investigation of corrosion compatibility of a 9Cr-1Mo disc sample (P91) exposed to Pb-17Li in the above test facility. The experiment was carried out at 923K for 2000 h continuously. The rotational speed of the disc was set at 700 RPM which corresponded to a range of linear velocities from 0.18 m/s to 1.4 m/s along the sample radii. After completion of the experiment, the sample was cleaned off the adhering lead-lithium and weight loss due to corrosion was recorded. The effect of corrosion on microstructure was investigated on the cross section at different positions along the sample radii using SEM-EDS. The composition of the exposed lead-lithium eutectic was also analyzed through ICP-AES. The results obtained from the above characterization are discussed in this paper.

P1-074 **Tritium Transport calculations for the IFMIF Tritium Release Test Module**

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The objective of the IFMIF Tritium Release Test Module (TRTM) is to measure online the tritium release from breeder ceramics and beryllium pebble beds under high energy neutron irradiation. Tritium produced in the pebble bed of TRTM is swept out continuously by a purge gas flow, but can also permeate into the module's metal structures, and can be lost by permeation to the environment. Accordingly analyses on the tritium inventory are performed to support IFMIF plant safety studies, and to support the experiment planning. This paper describes the necessary elements for calculation of the tritium transport in the Tritium Release Test Module as follows: (i) applied equations for the tritium balance, (ii) material data from literature and (iii) the topological models or the computation of the five different cases; namely release of tritium from the breeder solid material into the purge gas, loss of tritium over the capsule wall, rig hull, container wall and purge gas return line in detail. The problem of tritium transport in the TRTM has been studied and analyzed by the Tritium Migration Analysis Program (TMAP) and the adapted Fusion Devoted-Tritium Permeation Code (FUS-TPC). TMAP has been developed at INEEL and now exists in Version 7. FUS-TPC Code was written in MATLAB with the original purpose to study the tritium transport in Helium Cooled Lead Lithium (HCLL) blanket and in a later version the Helium Cooled Pebble Bed (HCPB) blanket. This code has been further modified to be applicable to the TRTM. Results from the computations for the five computational cases concerning safety and experiment analyses are presented.

P1-075 **IFMIF-LIPAc beam diagnostics and its challenges**

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The International Fusion Materials Irradiation Facility (IFMIF) aims at providing a very intense neutron source (10**17 neutron/s) to test the structure materials for the future fusion reactors, beyond ITER (International Thermonuclear Experimental Reactor). Such a source will be driven using 2 deuteron accelerators 125 mA cw up to 40 MeV impinging into a lithium liquid curtain, thus producing very high neutron flux with a similar spectrum as those expected in fusion reactors. A validation phase was decided for this 10 MW facility consisting partly in the design of the prototype accelerator LIPAc (Linear IFMIF Prototype Accelerator). LIPAc facility, currently under construction at Rokkasho (Japan), will accelerate a 125 mA cw and 9 MeV deuteron beam up to the first superconductive linac module (4 for IFMIF). LIPAc beam diagnostics are used as scaled prototypes for the IFMIF accelerator ones. Here, a description of the proposed IFMIF accelerator beam instrumentation and diagnostics, techniques and current layout, together with the challenges encountered will be presented.

P1-076 **Microstructural characteristics of commercial W-1% La_2O_3 alloys**

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Tungsten is considered as a candidate material for application as plasma-facing structure in fusion reactors. However, tungsten has a high ductile to brittle transition temperature (DBTT), and a low recrystallization temperature, which are the most critical restrictions on its application. Since La_2O_3 dispersion can increase both the recrystallization temperature and machinability of tungsten, W–1% La_2O_3 alloy is recommended to be plasma-facing material for ITER. The investigation of microstructural characteristics of W–1% La_2O_3 alloy is very important to understand its DBTT and to evaluate its applicability in ITER. The microstructure of commercial W–1% La_2O_3 alloys with different states has been investigated through optical microscopy, scanning electron microscopy, and transmission electron microscopy combined with electron diffraction and energy-dispersive X-ray analyses. The W–1% La_2O_3 alloy as rolled or forged gets a superior compactness with no pores and cracks, and has plenty of particles dispersed in the matrix compared with commercial pure tungsten as sintered or rolled. Several kinds of phases including lanthanum oxide, La_2O_3, tungsten oxides, WO_3 and W_3O_8, as well as lanthanum tungsten oxide, $La_2(WO_4)_3$, were identified to coexist in the W–1% La_2O_3 alloys. The La_2O_3 particles have a shuttle-like morphology and a mean size up to 1.3 μm in short axis and 4.7 μm in long axis, which tend to be the most unfavorable factors in the mechanical performance of the alloys. Ellipse-like WO_3, polygon-like W_3O_8, and ellipse-like $La_2(WO_4)_3$ particles have a mean diameter of about 400, 650, 220 nm, respectively. Formation of WO_3, W_3O_8, and $La_2(WO_4)_3$ phases were discussed, suggesting that the formation was closely related to a seriously inadequate hydrogen reduction during the alloy preparation procedures. Amorphous phase with an uncertain shape was found to be present in the W–1% La_2O_3 alloys. The ductility of the W–1% La_2O_3 alloys was also discussed.

P1-077 **Study on the capsule material feasibility in ITER environment**

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A neutron activation system (NAS) measures neutron fluence at the first wall and the total neutron flux from the plasma, providing the fusion power evaluation. A pneumatic transfer method is conventionally utilized to transfer encapsulated activation samples between the irradiation stations and counting station. The temperature of the irradiation stations near the first wall could reach too high for the conventional polymer-based materials, such as polyethylene, to be used as a NAS capsule. Preliminary investigation is conducted for various candidate materials in terms of their operational feasibility such as thermal and nuclear characteristics. Carbon Fiber-reinforced Carbon (CFC) is chosen as a promising candidate by its high thermal resistance and low neutron activation. Various shapes of Capsules are fabricated by using CFC with different fabric structures. With the fabricated CFC capsules, repetitive pneumatic transfer experiments were performed to evaluate its impact resistance and lifetime, by measuring maximum endurable transfer speed of capsule without damage and the number of capsule transfer in endurable transfer speed, respectively. The experimental results as well as the preliminary investigation show the feasibility of using CFC as a capsule material, suggesting the pneumatic operational conditions of NAS.

P1-078 **Influences of alloying elements and tempering on the impact and creep properties of Korean RAFM steel**

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Reduced activation ferritic-martensitic (RAFM) steel is considered a primary candidate for the structural material in a fusion reactor. The operational design window for a blanket is limited by the high-temperature creep and low-temperature irradiation embrittlement of the structural material and therefore it is essential to develop the RAFM steel that is able to withstand high temperatures and high energy neutron irradiation. The present work aims at developing advanced RAFM steel with the better creep and irradiation resistance. Various 9Cr-based FM steels were designed, where the amounts of alloying elements vary systematically to control the chemistry and distribution of the precipitates, and to improve the grain boundary characteristics. The program alloys were produced by vacuum induction melting and then formed into a plate by hot rolling, which is followed by various austenitization and tempering treatments. A series of mechanical tests was carried out to determine the tensile, impact, and creep properties of tempered plates, based on the results of which several promising alloys were selected for further investigation. The Charpy impact test reveals that the addition of a trace amount of Zr improves the impact resistance with a slight sacrifice of creep strength for a given tempering condition. In addition, the ductile brittle transition temperature (DBTT) of Zr-added alloy is less sensitive to the tempering conditions, whereas the short-term creep resistance deteriorates with the tempering time, which leads to a process modification with a reduction of the degree of tempering. It is confirmed that the under-tempered Zr-added alloy exhibits a remarkable enhancement in the creep resistance while the DBTT of the alloy is comparable to those of conventional RAFM steel.

P1-079 **Impact Properties of Electron Beam Welds of V-4Ti-4Cr alloys NIFS-HEAT-2 and CEA-J57**

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Development of joining techniques is a key technological issue for successful application of V/Li blanket concept of fusion reactor. Electron Beam (EB) welding can provide very narrow zones of weld joints and as a result the mechanical properties can be improved in comparison with GTA and laser welding. The aim of this work was to determine the impact properties of EB welds of V-4Ti-4Cr alloys of NIFS-HEAT-2 (NH2) and CEA-J57 (J57) grades. Samples were made by bead-on-plate EB welding in high vacuum atmosphere in SFC Co., Ltd. (Yokohama, Japan). Optimal welding power of 1.5 and 2.55 kW was determined experimentally and used for 4 and 7 mm thick plates of NH2 and J57 respectively. Charpy 1.3 size specimens were machined with V-notch placed in the Weld Metal (WM) perpendicular to the rolling direction. Coarse-grained columnar crystallites were formed in WM. Thickness of WM does not exceed 1 mm. Hardness of WM for both grades averaged 180 kg/mm2 while hardness of Base Metal (BM) is about 135 kg/mm2. It is caused by decomposition of Ti-CON precipitates in WM followed by the solid-solution hardening of V-matrix with O. Precipitation bands (Ti-CON), naturally presented in V-alloys, are decomposed partially in the Heat Affected Zone (HAZ) adjoining WM. Absorbed energies of WM for NH2 and J57 were near similar and ranged from 12 to 15 J in the temperature interval of test 125÷290 K. At 77 K the absorbed energy of J57 alloy decreased sharply (~1.80 J) while that one of NH2 remains high enough (10.43 J) indicating correspondingly brittle and ductile character of fracture which were confirmed by fractographycal examinations.

P1-080 **Evaluation on Defect in the Weld of Stainless Steel Materials using Nondestructive Technique**

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The objective of this study is to evaluate the elastic waves characteristic on the crack in the welds of stainless steel materials using guided wave and acoustic emission, nondestructive tests. The stainless steel is expected as candidate of structural piping material under high temperature condition in nuclear fusion instrument, and a tungsten inert gas (TIG) weld technique was applied for making its jointing. The various defects sizes of 3mm, 5mm, 10mm, 15mm and 20mm were induced in the weld material. The guided wave, one of elastic waves, can propagate very long pipe, and a mode of guided wave easily changes to lots of modes by the defects in the structure. By the analysis of the relationship between the mode conversion and the defects we can evaluate the defects in weld material. In present study Nd-YAG laser was used to excite the guided wave non-contact method, and AE technique was also used to clary the mode conversion of guided wave by defect because lots of AE parameters of energy, count and amplitude can give more chances for analysis of mode conversion. AE parameters were changed by not only the defects in the weld material but also the distance of guided wave source and AE sensor. The optimal AE parameters for the evaluation of the defects in weld using laser guided wave were derived.

P1-081 **Nanoindentation by using CSM as tool to measure changes in mechanical properties on RAFM steels irradiated with heavy ions.**

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High energy fusion neutrons produce atomic displacement damage and nuclear transmutations atoms (H, He) within the irradiated materials. From the point of view of structural materials, these microstructural changes will induce severe mechanical properties degradation, leading to strong hardening and/or embrittlement effects.

Within the Fusion Materials Community one of the major scientific challenges related with the structural steels is to obtain a deep knowledge of the interactions between irradiation damages (dpa) and gas atoms (He and H) and their effects on the microstructural and mechanical properties. The present work presents a systematic study on RAFM steels (EUROFER'97 and EU-ODS-EUROFER) by using nanoindentation with the module of continuous stiffness measurement. In the fusion radiation environment neutron damages up to 150 dpa and ~10appmHe/dpa and ~45 appmh/dpa are expected. Due to the lack of facilities to emulate the fusion environment, it is necessary to emulate the effect of the irradiation damage somehow. Irradiation with self-ions has been carried at room temperature for different amounts of damage, in order to simulate neutron damage. The average damage caused has been 0.05, 0.2, 6.5 and 10 dpa, covering a relative large damage spectrum and the specimen temperature was controlled at every moment of the irradiation procedure with thermocouple and thermographic. The stopping ionic range and the large cross-section of the Fe-ions produce an extremely shallow layer of damaged material. Its thickness is of a few microns and depends on the irradiation energy. This characteristic configuration makes nanoindentation the optimal tool to get the information about changes on mechanical properties of these complex and metallurgical new materials, by means of CSM. This module analyses the properties as a function of indenter depth, so is relatively easy to evaluate how hardness and stiffness change by comparing the behaviour of the damaged layer with that of the substrate in the as-received condition. The results obtained correlate the hardening of the steel with the damage level, showing that hardness and stiffness increase as a function of depth. This work validates and enforces the use of nanoindentation as a technique to study irradiated materials.

P1-082 **Mechanical properties of nano-particle dispersion stren gthened V-4Cr-4Ti alloy**

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In an attractive design of liquid Li blanket for fusion reactors, V-4Cr-4Ti alloy is assigned as the first candidate structral material. In the service environment, this alloy is expected to sustain high load at elevated temperature, thus its creep deformation is of concern.
Rencently, nanoparticle dispersion strengthening has been widely used to improve the creep resistance of fusion structural materials including vanadium alloys. It showed that the creep strength was efficiently increased by the large number density of nanoparticles. However, the grain sizes were reduced by the mechanical alloying (MA), which usually leads to degradation of creep property.
In order to improve the creep properties of MA-fabricated vanadium alloys further, it is necessary to increase their grain sizes. This can be achieved by applying high-temperature annealing. On the other hand, high-temperature annealing usually makes the particles coarsen, and thus reduces the number density of the particles, softening the alloy. Possible way to obtain coarsened grain size without remarkable decrease in number density of particles is to introduce the particles which have high thermal stability.
This study uses MA to fabricate yittrium and carbides added V-4Cr-4Ti alloys. High-temperature annealing is carried out to modify the grain size of these alloys. Hardenss, tensile strength and creep properties have been tested. Microstructures of the MA-fabricated V-4Cr-4Ti alloys with yittrium and carbide additions are characterized with a transmission electron microscope (TEM). The effects of grain size and thermal stability of nanoparticles on the creep properties of V-4Cr-4Ti alloys are discussed.

P1-083 **Joining Technologies of RAFM steel CLF-1 for Fabrication of ITER Test Blanket Module**

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Reduced activation ferritic/martensitic steels (RAFMs) are recognized as the primary candidate structural materials for fusion blanket systems. Now a type of RAFM steel CLF-1 is planned to be used as the structural materials for Chinese helium-cooled/solid ceramic breeder ITER test blanket modules (CN HCSB TBM). For the fabrication of CN HCCB TBM, joining is an inevitable procedure; however, the joining characteristics of CLF-1 steel are still a critical issue due to its self-hardening characteristics. Several joining processes are chosen as the fabrication techniques according to the design of CN HCCB TBM, including hot isostatic pressing (HIP), electron beam welding (EB) and tungsten inert gas welding (TIG). The feasibility of CLF-1 for the fabrication of TBM was investigated in this paper. Plates and tubes of CLF-1 steel with different size are used for the study of different joining technologies. By mechanical properties test and microstructure characterize, both of the TIG and EB welding showed that preheating before welding was not necessary but a post-welding heat treatment (PWHT) at 1013 K for 3h is required to improve the mechanical properties of the welds. After PWHT, the welds showed fully tempered martensitic structure but a coarsening of grain size was detected in the weld zones, induced a shift of Ductile to Brittle transition temperature (DBTT). The experiment results show that to get high quality HIP joints, the parameters should be chosen as lower than 3.2 µm surface roughness and HIPed at 150MPa for 2h around 1323 K. Currently these joining technologies are being carried out on the different size plates and on the joining for the plates and pipes to make the joining techniques applicable for TBM manufacturing.

P1-084 **Performance characterization of the FLEX low pressure helium facility for fusion technology experiments**

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FLEX (Fluid Dynamics Experimental Facility) is a multi-purpose small scale gas loop for research on fluid- and thermodynamic investigations, especially heat transfer, flow field measurements and gas purification. Initially built for investigation on mini-channel gasflow in the framework of IFMIF (ITHEX - IFMIF Thermo-Hydraulic Experiments), the loop has been continuously extended making it valuable for further experimental campaigns, e.g. to measure the pressure drop in helium cooled pebble beds (HCPB).

The main parameters of the loop that can be operated with several pure gases like Helium, Nitrogen, Argon or air in a closed loop setup from sub-atmospheric to medium overpressure, or alternatively open to atmosphere are:

-Operation gas pressure 0.02 – 0.38 MPa abs.

-Test section pressure head up to 0.12 MPa

-Tolerable gas temperature RT-200°C (up to 500°C at low flow rates)

-Mass flow 0.2 – 12 g/s for Helium (depends on test section)

Apart from the usual loop instrumentation, the facility offers extensive instrumentation for the test section. Currently, 40 thermocouples, 24 differential pressures and 8 absolute pressures can be measured, 8 channels of bridge-amplifiers (e.g. for strain measurement) are available. Heating power can be supplied by 4 separate controllable 3kW DC electrical power supplies. Various flow field measurement techniques such as hot wire anemometry (CTA) and particle image velocimetry (PIV) are also available in the laboratory.

This paper will give a detailed view of the loop assembly with its components to generate and regulate the massflow and loop pressure. The measurement instrumentation will be presented as well as a representative massflow-pressure drop characteristic. Furthermore, the achievable gas tightness and degradation of the gas purity will be discussed.

P1-086 **Engineering Design of the IFMIF EVEDA Reference Test Cell and Key Components**

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The IFMIF (International Fusion Material Irradiation Facility) test cell (TC) is the enclosure to accommodate the lithium target assembly and all of the test modules to perform intensive neutron irradiation on material specimens. As the meeting point of the major systems of IFMIF, the TC is required to provide sufficient neutron and gamma shielding to adjacent rooms and cells, controlled internal environments, and reliable and conveniently-maintainable interfaces with other systems. During the IFMIF–EVEDA (Engineering Validation and Engineering Design Activities) phase an optimized TC design has been introduced, developed and considered as the reference TC. In this paper, the engineering design of the IFMIF-EVEDA reference TC is described with emphasis on the latest updates of the following key components:

1) Active cooling system of the biological shielding. Buried pipes are selected for active cooling of the TC biological shielding as well as the closed liner which is cladded on complete internal surfaces of the TC. Technical features and layout of the cooling pipes are preliminary defined based on neutronic and thermal hydraulic calculations.

2) TC leak tightness solution. Newly developed TC leak tightness solution features independent TC covering plate, rubber based top sealing gasket, and welding seams of the interface shielding plugs. Removable top shielding materials are excluded from the TC sealing function.

3) Piping and cabling inside and around the TC. Engineering design of the piping and cabling plugs as well as the arrangement of pipes /cables from the test modules under the TC covering plate and the access cell (AC) floor are described in detail. Locations of pipe and cable penetrations between the TC and AC are carefully selected for convenient access and maintenance.

P1-087 **Preliminary design of the Neutron Spectral Shifter dedicated to the IFMIF Liquid Breeder Validation Module**

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The International Fusion Materials Irradiation Facility (IFMIF) is a D–Li stripping type neutron source with the purpose of providing typical fusion irradiation conditions for material testing. The Liquid Breeder Validation Module (LBVM) is one of the medium flux test modules of IFMIF. The objective of this module is to deal, in a versatile set up, with some of the Liquid Breeder Blankets R&D needs which require a relevant neutron environment under the temperature operational conditions expected in the DEMO reactor. Previous analyses have shown that the main irradiation parameters (dpa, He/dpa and H/dpa) in the medium flux area of IFMIF could be improved to fit to the expected ones in DEMO reactor for functional materials of liquid breeder blankets. Therefore, the design of a Neutron Spectral Shifter (NSS) has been considered to optimize the irradiation conditions of the LBVM experiments.

The present paper summarizes the work devoted to the design of the LBVM Neutron Spectral Shifter. The proposed concept essentially consists of a steel structure where tungsten plates, working as shifter material, are supported. This design enables to fulfill the neutronic requirements at the same time as the cooling capability and the mechanical integrity of the different components are assured. The NSS is expected to be irradiated during one-year irradiation campaign and thus a high damage value is foreseen. For this reason, a great effort has been performed to find the suitable design code which takes into account the material degradation appearing under irradiation environments. To achieve this target, the results from the mechanical analysis of this device have been categorized and compared to an associated limiting value, according to the basis defined in the Structural Design Criteria for ITER In-vessel Components (SDC-IC).

In this paper, the results of the neutronic, thermo-hydraulic and mechanical simulations of the NSS carried out to validate the design are described.

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P1-088 **IFMIF-EVEDA SRF Linac Couplers Test Bench**

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The IFMIF-EVEDA SRF Linac is a cryomodule equipped with eight superconducting HWR cavities, operating at the frequency of 175 MHz and powered by 200kW CW RF couplers. Before assembling the couplers to the cryomodule, it is necessary to process them using high levels of RF power. In order to perform this conditioning, the power couplers must be connected to a RF network which is fed by an RF source and ended with a load or a short-circuit, depending on the conditioning mode to be applied. A test bench has been designed for the conditioning of the SRF LINAC couplers. The main component is the "test box", a resonant cavity where two couplers will be assembled to transmit the 200 kW from the RF source to the appropriate termination. The test box includes a large pumping port allowing an efficient pumping of the entire vacuum volume limited by the coupler ceramic windows. Several diagnostics as light detectors, vacuum gauges and thermal transducers will provide information on the relevant parameters for the control of the RF conditioning process. In addition, a support frame has been designed to maintain the whole assembly and reduce the mechanical stress on the couplers.

P1-089 **Electrical insulating radiation-resistant coatings for the design elements of ITER Blanket**

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OJSC NIKIET investigates (experience is ~50 years) the creating of technological processes for the plasma sprayed coatings with various functional intention (electrical insulation, thermal protection, neutron absorption, tribology, corrosion resistance). These coatings have been developed in order to provide the ability to operation of nuclear power reactors and investigation facilities. The investigation works include the following stages:

-selection of coating material (mainly as powder);

-validation of coating ability to operation;

-development of plasma spraying technology on design elements;

-fabrication of experimental and industrial batches of coated products. Predominantly the investigations are devoted to the development of plasma-sprayed electrical insulating radiation-resistant coatings. This paper carries material on the following investigations which have been performed during 2011 - 2013:

1. The comparative analysis of technologies (plasma spraying and detonation spraying) to create the electrical insulating coatings.

2. The results of static strength testing for the full-scale products with electrical insulating coatings. The using of thermal non-destructive method for the coating quality evaluation.

3. The modifying of plasma-sprayed coatings in order to decrease the friction coefficient.

P1-090 **Metallurgical Analysis of Lithium Test Assembly Operated for 1200 hours**

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Lithium free-surface flow experiments to verify the design of the IFMIF target have been carried out at Osaka University, Japan. The lithium test assembly which is the free surface test section is the 1/3 scales of the IFMIF lithium target, and consists of a double reducer nozzle and an opened straight-horizontal flow channel, made by type 304 ss. The maximum velocity is 15 m/s at the outlet of the nozzle. The lithium test assembly has been replaced to new one in 2010 after 1200 hours of total operation time at 300 ºC. One key issue in the development of the IFMIF is the corrosion/erosion of the lithium components. It is thought that the replaced test assembly is important to understand the corrosion/erosion behavior as the demonstration experimental data. This paper describes the metallurgical analysis of the replaced assembly. Slight irregularities which were trace of high-speed lithium flow were observed at the tip of the nozzle. On the other hand, mottled unevenness with many micro-cracks of a few micrometer depths was observed at the inlet of the nozzle, whose velocity ratio was 0.1 - 0.4 as compared with the nozzle tip. In this region, a thin altered layer due to the carbide formation was detected from the observation of the cross section. Since the hardness of the altered layer was increased as compared with that of the base metal, it was considered the micro-cracks were nucleated by the thermal transient at the start / stop operation of the lithium test loop. The slight irregularities observed at the tip of the nozzle might be the results of exfoliating of the altered layer by high-speed lithium flow. From the metallurgical analysis, it was newly proven that carbon control in lithium was also important for corrosion / erosion protection of the IFMIF components.

Topic E Exvessel

P1-091 **Pendulum Support of Plasma Vessel W7-X**

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The world's largest superconducting helical advanced stellarator Wendelstein 7-X (W7-X) is under construction at the Max-Planck-Institut für Plasmaphysik (IPP) in Greifswald/Germany. Its ultimate goal is to verify the stellarator magnetic confinement concept as a viable option for a demonstration fusion power-plant. The three dimensional shaped toroidal plasma (major diameter of 11 m) is enclosed by a Plasma Vessel of similar complex shape. The Plasma Vessel and the Outer Vessel generate together with the Ports the cryostat to achieve a cryo vacuum for the superconducting magnet system, operating at about four Kelvin. Due to adjustment and thermal movements the Plasma Vessel has to be supported vertically adjustable and horizontally movable. This will be carried out by special Plasma Vessel Supports. The individual Plasma Vessel modules are supported by special sliding tables during assembly to reach stability but allow also small movements for welding shrinkage. After the assembly phase of W7-X the Plasma Vessel will be supported by pendulum supports and additional adjustable horizontal supports.

The paper will give an overview about development and status of Plasma Vessel Supports. The challenging transition from the temporarily support during the assembly to the final supports will be explained. Critical aspects and associated problems of design, tests, manufacturing and assembly will be described also.

P1-092 **Thermal-hydraulic analysis for ITER Upper ELM Coil**

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A thermal-hydraulic analysis has been performed for the design of ITER Upper edge localized modes (ELM) coil using ANSYS. The global parametric finite element model based on its preliminary design structure has been established in ANSYS. The thermal loads include the ohmic heat from the working current and the nuclear heat from the plasma. The cooling water flowing through the coil and its pressure onto the conductor inner wall are modeled to provide the cooling power. The radiation from the environment is also considered to cover all thermal issues. The cooling scheme and the water flow velocity are optimized and its impact on the structure temperature distribution and pressure drop are checked to find out the best plan.

P1-093 **Intelligent Controller of aFlexible Hyprid Robot Machine for ITER Assembly and Maintenance**

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The assembly and maintenance of International Thermonuclear Experimental Reactor (ITER) vacuum vessel (VV) is highly challenging since the tasks performed by the robot involve welding, material handling, and machine cutting from inside the VV. To fulfit the tasks in ITER application , this paper presents a hybrid redundant manipulator with four DOFs provided by serial kinematic axes and six DOFS by parallel mechanism.

Parallel robots exhibit good performance in terms of rigidity, accuracy, and dynamic characteristics. Despite these advantages, modeling and control of flexible manipulators is more difficult than controlling rigid manipulators. Thus, in machining, to achieve greater end-effector trajectory tracking accuracy for surface quality, a robust control of the actuators for the flexible link has to be deduced. In this paper, we have investigated the intelligent control of a hydraulically driven parallel robot based on the dynamic model and two control schemes have been developed: (1) Fuzzy-PID self tuning controller composed of the conventional PID control and with Fuzzy logic; (2) Adaptive neuro-fuzzy inference system-PID (ANFIS-PID) self tuning of the gains of the PID controller. The obtained results confirm the theoretical findings, i.e., the Fuzzy–PID and ANFIS-PID self tuning controller can reduce more tracking errors than the conventional PID controller. Amongst these methods, ANFIS has provided the best results for controlling robotic manipulators as compared to the conventional control strategies. Subsequently, The serial component of the hybrid robot can be analyzed using the equilibrium of reaction forces at the universal joint connections of the hexa-element. To achieve precise positional control of the end effector for maximum precision machining, the hydraulic cylinder should be controlled to hold the hexa-element in a fixed position with minimal positional error being transmitted to the parallel linkages.

Finally, simulated results that demonstrate the robot behaviors are presented.

P1-094 **Design of the Tore Supra West divertor structure according to nuclear construction code**

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Actively cooled tungsten plasma facing components will be used in the ITER divertor. In order to fully validate such a technology (industrial manufacturing, operation with long plasma duration), the implementation of a tungsten axisymmetric divertor in the tokamak Tore-Supra is studied. With this major upgrade, so called WEST (Tungsten Environment in Steady state), Tore-Supra will be the only European tokamak able to address the problematic of long plasma discharges with a metallic divertor. The paper will present the design studies implemented in 2012 to ensure compliance with RCCMR design criteria even if they are not required for this type of experimental fusion machine.

First, 3D electromagnetic analyses have been performed to calculate the various loadings : eddy currents, the vertical halo forces and the electromagnetic forces on the coils and passive structure during disruption and VDE. Axisymmetric loads are estimated by a 3D analysis on a 20-degree machine sector. Furthermore, a steady state electrical calculation of half of the structure is performed to evaluate the sideway force and the tilting moment created by the asymmetry of the toroïdal halo current plasma in the structure.

The whole divertor structure is submitted to huge mechanical loads in normal operating conditions and especially during disruption events. The validation of the mechanical integrity of this global structure is ensured by the respect of RCCMR stress criteria. Various conservative cases of static loadings are to be considered in order to represent all the transients. 3D static elastic thermo mechanical analyses, performed with the ANSYS code, give the stress level in the structure. In addition, some local studies have also been done: on singularity zones and on bolting junctions for example.

P1-095 **Thermal Analysis on Detailed 3D Finite Element Models of ITER Thermal Shield**

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Thermal Shield (TS) is to be installed in the ITER tokamak in order to cut off the radiation heat load transferring to the magnets operating at 4.5 K. The helium coolant flows inside the cooling tube attached on the panel surface with inlet temperature 80 K. The TS shall be properly designed so that the radiation heat load is below the specified requirement for the magnet. This paper describes how the TS is modeled in a thermal point of view and shows typical results from the thermal analysis. The 3D finite element models of 40 degree sector of Vacuum Vessel Thermal Shield (VVTS) and 20 degree sector of Cryostat Thermal Shield (CTS) have been developed in this study. The simplified model of cooling tube is validated by its local thermal analysis. The passively cooled removable panel on the CTS for the in-cryostat maintenance of the magnet is modeled in detail. Thermal contact conductance is assigned at the joint between the removable panel and the frame. The thermal contact conductance is obtained from the analysis of local TS model with the design requirement of temperature rise at the joint. The axial conduction through the CTS support is included in the model with the predetermined support end temperature, which is obtained from the analysis of the cryostat connected to the CTS. Labyrinth geometry at the edge of CTS is separately considered in the analysis. Surface to surface radiation in the labyrinth is analyzed for passive cooling design. Design improvement is proposed at the specific location of CTS where the peak temperature is above design criteria. The total heat load from the TS to the magnet is evaluated from the analysis. Finally, a parametric study is performed to investigate the sensitivity of coolant inlet temperature on the heat load to the magnet.

P1-096 **Fabrication Results of Full Scale Mock-up for ITER VV Port in Korea**

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Main purpose of the ITER Vacuum Vessel (VV) port is to give penetration from VV to cryostat for plasma diagnostic, plasma heating and maintenance of in-vessel component, etc. Korea has responsibility for procuring of all major ports at the equatorial & lower level and other components including the VV supports, Neutral Beam (NB) duct liners, and port sealing flanges. The procurement arrangement was signed with the ITER Organization in late 2008. Fabrication contract for the main port components was signed with Hyundai Heavy Industries Co., LTD (HHI) in early 2010. After contract with HHI, the fabrication preparation activities had been performed. The fabrication preparation activities are composed of fabrication feasibility studies, preparation of fabrication drawings, material procurement, engineering analyses, several qualifications for major fabrication procedures and R&D. The real product fabrication was started in Nov. 2012 for the Lower Port Stub Extension (PSE). In particular, a full scale mock-up of the Lower PSE #16 has been fabricated to minimize all expectable risks which can occur during real production due to tight tolerances and strict inspection required from both ITER technical requirements and the RCC-MR design/construction rules. Another aim is to verify fabrication feasibility and sequence of the port. The main reason for selection of the Lower PSE #16 is complicated double shell structure including the Divertor Rail Support with very tight dimensional tolerance requirement. Weldability, applicability of Nondestructive Examination (NDE) and welding distortion have been evaluated during the full scale mock-up fabrication. The performance of welding jig was also major concern. The fabrication sequence of real product has been established based on the full scale mock-up fabrication results. In this paper, major technical results of full scale mock-up fabrication will be presented related on main fabrication procedure, welding, NDE and 3D dimensional inspection. After that, progress of real product fabrication activities are will be introduced.

P1-097 **Bolted Ribs Analysis for the ITER Vacuum Vessel using Finite Element Submodelling Techniques**

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The ITER Vacuum Vessel (VV) primary function is to enclose the plasmas produced by the ITER Tokamak. Since it acts as the first radiological barrier of the plasma, it is classified as a class 2 welded box structure with a support and leak tightness role, according to RCC-MR 2007. Its final mechanical design has to be approved by the *Autorité de Sureté Nucléaire*. The VV is made of an inner and an outer D-shape, 60 mm-thick double shell connected through thick massive bars (housings), toroidal and poloidal structural stiffening ribs. In order to provide neutronic shielding to the ex-vessel components, the space between shells is filled with borated steel plates, called In-Wall Shielding (IWS) blocks, and water. In general, these blocks are connected to the IWS ribs connected to adjacent housings.

The development of a FE model of the ITER VV including all its components in detail is unaffordable from the computational point of view due to the large number of degrees of freedom it would require. This limitation can be overcome by using submodelling techniques to simulate the behaviour of the bolted ribs assemblies. Submodelling is a FE technique which allows getting more accurate results in a given region of a coarse model by generating an independent, more finely meshed model of the region under study. Results of the coarse model in the cut boundaries are used as boundary conditions of the submodel. The principle behind submodelling assumes that the cut boundaries are far enough away from the stress concentration regions.

In this paper, the methodology and several simulations of the VV bolted ribs assemblies using submodelling techniques are presented. Analyses on detailed submodels of these regions have been developed, accounting both for the detailed geometry of their components and their interaction. A stress assessment has been performed for the elements involved in the assembly considering possible types of failure and including stress classification and categorization techniques to analyse the critical sections. In particular, studies to optimise the number of bolts on each specific VV area have been developed to assess the compliance with RCC-MR 2007 requirements. It is shown as well that the submodelling technique is a very powerful tool to avoid the current computational limitations.

P1-098 **Electrical Parameters for KTX vacuum vessel**

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KTX (Keda Torus eXperiment) is a reversed field pinch magnetic confinement device designed by USTC and ASIPP. Based on the platform of KTX, the suitable magnetic shape for future practical controlled thermal-nuclear fusion should be explored. And some important physics phenomena, for example, the self-organization behavior and single-screw state of plasma also are researched. The vacuum vessel of KTX is double-C structure for detachment and maintainance convenient. The eddy current on vacuum vessel caused by the variation of the magnetic field is very complicated due to the complex structure. A method is presented to solve the electrical parameters of the vacuum vessel in this paper. The method can be used for the vacuum vessel with arbitrary shapes, especially for the vessel without closed current in toroidal direction as a result of poloidal insulating gaps. The distribution of eddy current is different from the toroidal loop. The time constant of the vacuum vessel can be derived from the decay characteristics of the eddy current.

Keywords: KTX, reversed field pinch, vacuum vessel, eddy current, decay characteristics

P1-099 **Detailed Analysis of Eddy Currents in Wendelstein 7-X**

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At present the stellarator experiment Wendelstein 7-X (W7-X) is under construction at the Max Planck Institute for Plasma Physics in Greifswald, Germany. The toroidally fivefold symmetric magnetic field is created by 50 non-planar superconducting coils. 20 additional planar superconducting coils as well as normal conducting correction and trim coils increase the experimental flexibility. Fast changes of currents, either in the coils or in the plasma itself, lead to eddy currents in all surrounding conducting structures. Since the plasma vessel (PV) is completely exposed to the main field, large eddy currents develop there. Up to now their distribution and effects were estimated using a simplified PV geometry only, neglecting openings and asymmetries with respect to the field symmetry. For the decline of the magnetic field a single time constant has been assumed. However, this approach turned out to be not sufficiently accurate, and particularly it did not allow to identify maximum current densities resulting from asymmetries. The development of eddy currents is now analysed in more detail with an FE model created in MAXWELL 3D. The model comprises the “as built” geometry of the PV, including all 254 partly asymmetric port openings, and representations of the coil and plasma currents. The temporal development of the currents in the superconducting coils during a fast discharge, taking into account the non-linear behaviour of the dump resistors, was simulated with the network code SIMPLORER and is also included in the eddy current calculations. Application of this new model not only yields local concentrations of current densities and forces in the PV wall, but allows also to judge eddy current effects on components inside the vessel. The paper is focused on the description of the FE model, the analysis method, and on the observed results.

P1-100 **The ITER EC H&CD Upper Launcher: Seismic Analysis**

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The electron cyclotron heating and current drive (EC H&CD) upper launcher (UL) is an in-vessel component of the ITER tokamak machine devoted to inject localized high microwave power, in order to counteract plasma instabilities (e.g. MHD activity). The UL, with length close to 6 m, is fixed by a support flange in the port of the vacuum vessel as a cantilevered structure and the nominal gap between UL and port is 25 mm only. During an earthquake, accelerations generated by the seismic event cause oscillations of the structure which might be amplified in case of resonance with the natural frequencies of the UL. A seismic analysis is therefore required in order to check the resistance of the UL to the earthquakes. This paper shows the procedure used for the seismic analysis of the UL and results are given in terms of displacements and stresses. The severe earthquake defined in ITER speech as SL-2 seismic event was considered. The response spectrum method was used in the analysis and floor response spectra (plot of acceleration versus frequency) provided by ITER/F4E at the upper level of the tokamak were applied to the supports as load. A seismic analysis of the UL integrated in the port is also here reported. The natural frequencies of the UL are far from the frequencies of the peaks in the applied spectra, so no resonance condition occurs. The obtained displacements and stresses of the UL are relatively small. The maximum total displacement is lower than 2 mm and the maximum equivalent stress is below 30 MPa. Since the highest excitation is the vertical one, most part of the total displacement is in the vertical direction. Afterwards, these results due to the seismic loads must be combined with displacements and stresses due to other loads affecting the UL such as the electromagnetic loads.

P1-101 **Design of ITER Vacuum Vessel In-wall Shielding**

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The ITER vacuum vessel is a torus-shaped, double wall structure. The space between the double walls of the VV is filled with in-wall shielding (IWS) and water. The main purpose of the IWS is to provide neutron shielding together with the blanket and VV shells during ITER plasma operation and to reduce the ripple of the Toroidal magnetic Field (TF). This paper presents the design of the IWS, considering loads, structural stresses and assembly feasibility, and also shows neutron shielding effect and TF ripple reduction caused by the IWS. According to the requirement of electrical resistance for the main vacuum vessel/ IWS combination, individual IWS blocks are installed on the double wall reinforcing ribs at a particular location with only several adjacent fixing points. Minimal gaps are maintained between the blocks to provide electrical isolation and enhance the assembly feasibility. To minimize the electromagnetic (EM) forces due to eddy currents, the IWS blocks are designed as a stack of thin plates.

After material investigations, Stainless Steel (SS) AISI 304B7 and SS 304B4 have been selected as primary IWS materials. 40 mm thick flat plates will be used in all areas and will fill 55% ~60 % of the volume between the vessel shells to provide an effective neutron shielding capability. The shielding material under the TF coil in the outboard part area has also the function of reducing the TF ripple in addition to providing the neutron shielding. This is achieved by using some shielding plates made of ferromagnetic material (FM) AISI SS430.

Electromagnetic analyses for both eddy current and magnetization have been performed, which determine the IWS design loads together with other related loads coming from the VV load specification. Detailed stress analysis has also been carried out for combined load cases for several typical FM blocks and non-FM blocks. The limit analysis is also performed when it is required. Pre-loading of the bolts and size of fasteners have been optimized according to the stress analysis results.

Topic F Neutronics

P1-102 **Current status of engineer design of KTX components**

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KTX device is one of the reversed field pinch (RFP) magnetic confinement device, it requires both toroidal and poloidal components for magnetic field induction. The former is provided by the toroidal field (TF) winding and the latter by the ohmic heating (OH) and the equilibrium field (EF) winding. The TF winding consists of 24 coils, OH winding and EF winding are included 20 coils and 16 coils, respectively. Besides those, KTX device are mainly composed of the vacuum vessel, conductive shell, saddle coils, support structure, drive structure as well. This paper is mainly dealt with the design of KTX device components, regarding to special key components, the calculation and analysis have been carried out. In addition, the assembly of KTX device have been considered.

P1-103 **Nuclear Analysis for ITER JA WCCB-TBM**

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In order to evaluate nuclear properties of the ITER JA WCCB-TBM (Water Cooled Ceramic Breeder Test Blanket Module) and ensure that the design conforms to the nuclear regulation for licensing, nuclear analyses have been performed for the WCCB-TBM including flame, shield, pipe-forest, bio-shield and AEU (Ancillary Equipment Unit). The WCCB-TBM is about 50 cm in width, 1700 cm in height and 60 cm in thickness. It is composed of Li2TiO3 pebble bed layer as the tritium breeder, beryllium pebble bed layer as the neutron multiplier, the low activation structural material F82H and cooling water. There are dog-leg gaps of 2 - 3 cm in width between the TBM and flame. Radial lengths of shield, pipe-forest, bio-shield and AEU are about 170 cm, 340 cm, 40 cm, and 790 cm, respectively. Nuclear analyses are performed with the Monte Carlo code MCNP5.14, activation code ACT-4 and Fusion Evaluated Nuclear Data Library FENDL-2.1. MCNP geometry input data of the TBM is created from CAD data with the automatic conversion code GEOMIT, and other geometry input data is created by manually. From the nuclear analyses, the following items are calculated: (1) neutron and gamma ray flux during DT operation, (2) neutron and gamma ray heating during DT operation, (3) tritium breeding ratio in the TBM, (4) neutron and gamma ray dose rate during DT operation, (5) rad-waste of each component, (6) decay heat after shutdown, (7) decay gamma-ray dose rate after shutdown. A special 'Direct 1-step Monte Carlo' method is adopted for the decay gamma-ray dose rate calculation. The activation calculations are carried out based on the SA2 operation scenario specified by ITER organization. By adopting the dog-leg gaps, decay gamma-ray dose rate can be drastically reduced and hands-on access is possible for shield. Detailed calculation results will be presented in this symposium.

P1-104 **Overview of neutronic analysis results for RF LLCB TBM**

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The results of breeding zone optimization from neutronics point of view for Russian conceptual design of lead-lithium ceramic breeder test blanket module (LLCB TBM) for ITER are presented. This investigation was performed as a part of basic design study for LLCB TBM. The main features of RF LLCB TBM are the following: dual coolant - LL in the breeding zone and helium in TBM structure; CB layers are in the TBM rear part; the flow of LL is organized in sequential-parallel system of poloidal channels with upward and downward flow. On the basis of 3-D neutronic analysis the optimal geometry and material composition of breeding zone have been proposed in order to provide for the maximum tritium production rate (TPR) without changing of TBM fixed overall dimensions and taking into account limitations on material's temperatures. Radial thickness of the first LL zone was increased up to 69 mm and total thickness of ceramics - up to 210 mm in comparison with the basic variant. Detailed distributions of total and specific nuclear heating and tritium production rates were estimated. Also the impact of various factors on tritium production rate in breeding zone has been investigated.

P1-105 **K-effective Benchmarking of SuperMC 2.0**

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Monte Carlo criticality calculations are performed routinely on large, complex models for reactor physics and criticality safety applications. Keff is the key parameter of criticality calculations. Super Monte Carlo Simulation Program (SuperMC) developed by FDS Team is a multi-functional simulation program mainly based on Monte Carlo method and advanced computer technology. SuperMC 2.0 can perform neutron, photon and coupled neutron-photon transport simulation. The main calculation method and benchmarking of keff in SuperMC 2.0 are presented in this paper. The eigenvalue equation of transport is solved by power iteration that is iterated simulation of particles batch-batch on computers. Hit estimators, absorption estimators and track length estimators were used to tally attribution to final keff results during the tracking of particles histories. Fission source distribution sampling method, convergence of source distribution and batch-batch simulation process were also introduced. To verify correctness of physical process and keff calculation method, a series of tests of pure fissionable nuclides critical sphere was carried out. Ten benchmarking examples from the International Handbook of Evaluated Criticality Safety Benchmark Experiments were adopted for comprehensive validation and verification.

The correctness of SuperMC2.0 was well proved by contrasting the results of SuperMC2.0 with the results of experiments and MCNP. SuperMC2.0 can be well used for criticality safety analysis of nuclear reactors.

P1-106 **FENDL-3 benchmark test with neutronics experiments related to fusion in Japan**

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A nuclear data library is one of the most important data which control the calculation accuracy in nuclear analyses for fusion reactor designs. IAEA compiles the best data from the evaluated nuclear data libraries in the world each nuclei for fusion reactor applications. This library is Fusion Evaluated Nuclear Data Library (FENDL) and the current version is FENDL-2.1. In 2008 IAEA started a new coordinate research project to update FENDL-2.1, to extend the neutron energy range from 20 MeV to 150 MeV and to include general purpose and activation data libraries for p- and d-induced reactions up to 150 MeV. This new library was named as FENDL-3. In 2012 the starter library release 4 of FENDL-3 was prepared in this project and it is just now being tested and the formal version will be released soon. We participate in the benchmark tests of the general purpose data library for neutron-induced reactions in FENDL-3 by using integral experiments with the DT neutron source at FNS in JAEA and TOF experiments with the DT neutron source at OKTAVIAN in Osaka University, and by using shielding experiments with the 40 and 60 MeV neutron sources at TIARA in JAEA. Here we present an important part of our benchmark test. The Monte Carlo code MCNP-5 and Sn code DORT or TORT were used for our benchmark test. The ACE and MATXS files of FENDL-3 supplied from IAEA Nuclear Data Section were adopted. We also used the current version FENDL-2.1 and the latest JENDL-4.0 for comparison in the experiments at FNS. As a result, it is found out that the starter library release 4 of FENDL-3 is as accurate as FENDL-2.1 or more. The detailed results with the formal release of FENDL-3 will be reported in the symposium.

P1-107 **Status of ITER TBM port plug conceptual design and analyses**

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One of the main engineering performance goals of ITER is to test and validate design concepts of tritium breeding blankets relevant to a power producing reactor. To accomplish these goals, three ITER equatorial ports are dedicated to the test of Test Blanket Modules (TBMs). These TBM sets are mechanically attached to the external Frame and form the whole TBM Port Plug (PP). The TBM PP consists of a TBM Frame and two TBM-sets (replaceable with Dummy TBMs). The Frame provides a standardized interface with the Vacuum Vessel/port structure and provides thermal isolation from the shield blanket. As one of the plasma-facing components, it shall withstand heat loads while at the same time provide adequate neutron shielding for the Vacuum Vessel and magnet coils, in the vicinity of Frame. The Frame design shall provide a stable engineering solution to hold TBM-Sets and also provide a mean for rapid remote handling (RH) replacement and refurbishment. This paper presents the conceptual design of TBM PP applied to Dummy TBMs, as follows.

- Description of the main features of the conceptual design for TBM PP with Dummy TBMs including interfaces, RH compatibility and materials choices.

- Assessment and optimisation of the TBM PP shielding capability in order to allow human access in the area behind the PP to perform hands-on maintenance operations; - Optimisation of the PP hydraulic performance in order to achieve the required pressure drops. Also heat transfer coefficient (HTC) is found to use as input data for thermal stress analysis of TBM PP.

- Electromagnetic (EM) and Structural Analysis: 3D numerical thermal-stress due to surface heat load (0.35MW/m2) and volumetric heat load, EM and stress analysis of the TBM PP are performed to evaluate static an cyclic strength of components from SDC-IC (or RCC MR).

P1-108 **Re-design of ITER GDC system based on a fixed electrode concept**

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ITER needs a reliable wall conditioning method to obtain a clean first wall for plasma breakdown after a venting. Direct Current GDC system is chosen as a baseline wall conditioning technique to reduce and control impurity and hydrogenic fuel out-gassing from plasma-facing components. Previously the design was based on a movable electrode integrated with In-Vessel-Viewing-System. However, this design was discarded due to its high risk of water leaks in flexible cooling pipes. GDC is now based on a fixed electrode concept. A review of the conceptual design of the GDC functions, safety, operation and maintenance has been successfully completed recently. Within the design boundaries constrained by operation conditions physics requirements have been defined on glow discharge current, anode surface and layouts. ITER GDC system design also needs to take safety and maintenance into account. Because of the staged installation strategy of ITER Temporary Electrodes (TE) will be used in the initial phases, while Permanent Electrodes (PE) will be employed once Port Plugs will be installed. TE with a relatively simple design will be used for early operation stages. PE will be integrated with Diagnostic Port Plugs (DPP), and the front surface will be in flush with the diagnostic first wall. This paper gives an overall description of the requirements and design implementation at the conceptual design level. The designs of the sub-systems are discussed against the corresponding requirements. The concept design of PE and TE is introduced in the paper.

Operation is discussed mainly in view of the ITER GDC I&C. In the ITER nuclear operation phase maintenance for its in-vessel components have to be carried out by Remote Handling (RH) in the Hot Cell Facilities. The concept design of the RH procedure is described. Important R&D results on topics such as, breakdown, glow homogeneity are listed as well.

P1-109 **The Electromagnetic design and Analysis for CFETR magnet system**

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CFETR superconducting tokamak is a national scientific research project of China, which stands for Chinese Fusion Engineering Testing Reactor. The plasma major radius R is 5.7 m and minor radius r is 1.6 m. The plasma center magnetic field strength (at R=5.7 m) is 5.0 T. The main goal of the project is to build a fusion engineering Tokamak reactor with its fusion power is 50~200 MW and should be self-sufficiency by blanket. The magnet system should fulfill the requirement of long pulse operation and high-beta plasma, as well as single-null plasma. Thus, the preliminary engineering design and computing for CFETR magnet system need do detail research works including the design of conductor, mechanical analysis and insulation design and so on. In this paper, the main design work was carried out including the electromagnetic analysis and parameters for magnet system. The conductor parameters ofthe magnetic coils will be discussed as detailed as possible. Besides, the safety requirement of the superconducting conductor such as the temperature margin and limiting current will be calculated. What is more, the engineering design work for CS and PF coils based on different diverters are analyzed in this paper.

P1-111 **Modeling and sizing of the heat exchangers of a new supercritical CO2 Brayton power cycle for energy conversion for fusion reactors**

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TECNO_FUS is a research program financed by the Spanish Government to develop technologies related to a dual-coolant (He/Pb-Li) breeding blanket design concept and to the auxiliary systems for a future power reactor (DEMO). One of the main issues of this program is the optimization of the heat recovery from the reactor and its subsequent conversion into electrical power. This paper is focused on the S-CO2 secondary power cycle proposed by the authors and, particularly, on the methodology employed for the design and the sizing of all the heat exchangers included in this cycle. Thus, a complete description of the eight heat exchangers will be presented, where four heat exchangers constitute the thermal source, three are recuperators and one is a precooler.

Printed Circuit Heat Exchangers (PCHE) are suggested in literature for its use, mainly due to the large pressure difference between the fluids and its compactness. Because of the complex behavior of CO2, their design is performed by a numerical discretization into sub-heat exchangers and implemented in the Engineering Equation Solver (EES) software. Different empirical correlations for the pressure drop and the Nusselt number have been considered and assessed. The CO2 critical point is nearly achieved in the precooler and in the low temperature recuperator (LTR) and therefore, their design also implies verification by numerical simulations using CFD, where the fluid properties have been programmed as a function of temperature. This work tries to estimate the size of the whole layout related to the heat exchangers involved in the power cycle. This will demonstrate the high compactness of this type of heat exchangers despite of the high values of effectiveness required by the power cycle.

P1-112 **Enhanced arrangement for recuperators in supercritical CO2 Brayton power cycle for energy conversion in fusion reactors**

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TECNO_FUS is a research program supported by the Spanish Government since 2009 which deeps into all the technological aspects involved in a specific concept of dual coolant (He/Pb-Li) breeding blanket called TBM (tritium blanket module). Each available heat source has a different thermal level which generates many problems in the coupling with the power cycle. In previous works authors have explored Brayton power cycles using both helium and supercritical CO2 as working fluids. Restrictions with the coupling have been overcome by different arrangements of combined cycles (helium Brayton/organic Rankine or supercritical CO2 Brayton/steam Rankine) and dual cycles, where the lower temperature heat source is used for a steam Rankine cycle and the rest of the available thermal energy is transferred to a supercritical CO2 Brayton cycle. Although these cycles provide high values of efficiency (more than 45%) the complexity of the layout is very high and the use of steam is required. As an alternative, a novel supercritical CO2 cycle was proposed by authors based on the inclusion of a new recuperator which heats a stream that bypasses the lower temperature heat source. In this work, a performance analysis of the novel cycle is carried out, explaining the interaction between different parameters and their effect on the efficiency. Special attention is paid to the interaction between the high temperature recuperator (HTR) and a new proposed one, where HTR is suppressed. This new arrangement includes the same number of elements of the classical supercritical CO2 Brayton cycle but replacing the HTR by the new recuperator which allows optimizing both the efficiency (values above 47% are achieved) and the balance of plant, avoiding the use of an extra recuperator.

P1-113 **Thermodynamic Evaluation on Power Conversion System Options for Potential K-DEMO Fusion Reactor**

Park, Il Woong[1]; Park, Min Young[1]; Park, Goon Cherl[1]; Lee, Young Seok[2]; Im, Ki Hak[2]; Kim, Hyoung Chan[2]; Kim, Kee Man[2]; Kim, Eung Soo[1]

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The major focus on this paper is to compare various coolants by evaluating their thermal-hydraulic performances for selection of the K-DEMO (Korean Fusion DEMO) coolant. This preliminary study evaluates pumping power and power conversion system efficiency of four different coolants - pressurized water, supercritical water, helium, carbon dioxide and supercritical carbon dioxide. The calculations have been conducted based on a plate-type blanket design, which was newly proposed by NFRI, for presumed operating conditions. Pressure drop and pumping power with total heat generation, operating pressure and channel width are calculated through fluid dynamics and thermodynamic analysis. The result shows that pressure drop is higher in the case of gas coolant than liquid coolant, but in the case of gas coolant, as the operating pressure rises or the channel width widens the pumping power decreases notably. Power conversion system efficiencies are evaluated using UNISIM code, a flow sheet analysis code. This study considered three different power conversion system options: steam Rankine, helium Brayton, and supercritical CO2 Brayton cycles. In addition, both direct and indirect coupling between blanket and power conversion systems are taken into account. The evaluation results shows that in case of water coolant pressure drop has a very small effect on system efficiency, but in case of gas coolant as pressure drop rises, the system efficiency decreases notably. However, in case of carbon dioxide coolant using direct cycle based on the supercritical CO2 Brayton cycle, special characteristics of supercritical carbon dioxide around critical point significantly reduced compression power of gas flow by changing the gas coolant incompressible. It eventually improved system efficiency notably even at moderate temperatures around 500C, and the effect of the pressure drop on the power conversion efficiency is significantly less than those of other gas coolants.

P1-114 **Assessment of Mesh-Coupled R2S Shutdown Dose Calculation Error Dependence on Voxel Resolution**

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In order to know when it is safe to perform maintenance upon fusion devices and future reactor designs the shutdown dose rate (SDR) arising from the decay of neutron-activated structural materials must be determined. Much work assessing SDRs is performed using rigorous two-step (R2S) methods, where determining activation from neutron irradiation and assessment of the gamma radiation arising from the decay of activated materials are split into two steps. These steps can be coupled by discretising the neutron flux across the geometry, activation and resulting gamma source into a common mesh of three-dimensional voxels. CCFE implements this method as the Mesh-Coupled R2S code MCR2S.

Ideally the finest resolution possible would be employed for the closest approximation to real behaviour, but memory constraints prevent practical use of voxel resolutions below 10cm for the large ITER & DEMO geometries currently of interest. In order to determine the optimal resolution for these SDR assessments and the magnitude and sign of error current resolutions might impose, a series of simulations were performed using the MCNP5 and MCR2S codes. The simulations were performed on simplified geometries allowing for smaller voxel sizes to be used, to search for a minimum resolution beyond which SDR estimates would plateau.

Resolutions were compared to neutron and gamma mean free paths within the geometry to determine the physical properties that could predict the optimal resolution for known material types, and thus determine errors on existing and future simulations. This is a relevant concern given the ongoing design work for ITER & DEMO for which SDR limits are a key safety factor. It was found that the best predictor of dose accuracy appeared to be mean free path of gamma rays produced by the activated materials- 2cm in ITER-like shielding. For resolutions lower than the gamma MFP SDR estimates plateaued, whilst for those ten or more times larger estimates routinely differed by a factor of two. The errors resulted in underprediction of SDR in in-vessel conditions and overprediction in locations behind shielding.

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Keywords: R2S, simulation, activation, MCR2S

Topic G Safety Issues

P1-115 **Activation analyses for the IFMIF-LBVM**

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The Liquid Breeder Validation Module (LBVM) will be one of the medium flux irradiation modules of the International Fusion Materials Irradiation Facility (IFMIF) neutron source. The objective of this module –presently under design- is the test of functional materials related to liquid breeders for future nuclear fusion power reactors (DEMO). This engineering design task has been performed taking into account the safety issues of the module following several months of continuous irradiation. This paper aims to describe the activation analyses performed to estimate the radioactive inventory, the decay heat and the expected contact dose from the activated materials of the module following an 11-month irradiation period. These calculations supply valuable information for different aspects related to the design of the module, such as the safety evaluation and the waste management and disassembly plan.

The neutron transport calculations to obtain the neutron spectra have been performed using the McDeLicious code. The ACAB nuclear inventory code, with the activation nuclear libraries EAF-2007, has been used for the activation analyses.

The main results point out that the contact dose of the LBVM materials (container, rigs and experimental capsules) is much higher than the hands-on-limits, as expected. Therefore, remote handling operations are requested for disassembling the module. It is important to remark that after 8 hours decay time, the contact dose rate of the LBVM decreases 70% for the EUROFER steel components (experimental capsules) and 50% for the 316 LN module container. Regarding the isotopic inventory, although the main activation comes from the module steel structures, the production of tritium and Po-210 in the lithium lead inside the experimental capsules deserved a careful analysis.

P1-119 **Busbar arcs at large Fusion Magnets: Conductor to Feeder Tube arcing Model Experiments with the LONGARC device**

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The inductive energy of about 40GJ stored inside the toroidal field (TF) coils of ITER provides a considerable potential of hazard in case of an accident. Calculations with the code system MAGS (MAGnet System) already proved that for most accidents the damage is limited to the coils themselves. However electric arcs moving along the power cables to the coils, the so-called busbars, may reach and penetrate the cryostat wall. Because of the lack of accurate numerical modeling and only sparse experimental data, small scale model experiments were initiated to investigate the propagation and destruction mechanisms of busbar arcs. The experiments are intended to support the development and validation of a numerical model. The whole campaign orientates very close to the expected ITER busbar arc scenario. An extrapolation of the calculation model to ITER scale shall avoid or at least reduce the need for possible full scale experiments during ITER licensing. If this cannot be achieved, the gained experimental experience will be helpful. This work shows the setup of the LONGARC device, taken into service in 2012. LONGARC represent a further step in arc model experiments at KIT. From basic, early experiments the sample conductors, their insulation and the geometrical arrangements of the model scenarios are successively brought closer to realistic ITER conditions. The present work continues the VACARC experiments that indicated that the inner feeder tube of a busbar won't withstand a powerful arc. The new 20m3 vacuum vessel allows carrying out the new experiments for larger geometric scale (1:3) with inner feeder tube as well for smaller scales (1:4 and 1:7), now also with outer feeder tube.

P1-120 **Preliminary results from a detritiation facility dedicated to soft housekeeping waste and tritium valorisation.**

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Nuclear waste management has to be taken into account for fusion machine using tritium as fuel. Soft housekeeping waste (e.g. gloves, tissues, protective clothes, etc.) is produced during the whole life as well as during the dismantling of the reactor and is contaminated by tritium under reduced (HT) and oxidized (HTO) forms. In collaboration with ENEA, a lab-scaled facility has been built at CEA Cadarache for soft housekeeping waste detritiation. The previously milled waste is placed in a reactor to be heated up to a temperature lower than the housekeeping melting point. A carrier gas is then injected in the detritiation reactor to remove tritium, thanks to the combined effects of temperature and carrier gas (type and feed flow). The tritiated gas exhausted from the detritiation reactor is then sent through a catalytic Pd-Ag membrane reactor (CMR) where tritium is recovered via isotopic exchange reaction and permeation phenomenon. The CMR consists of two parts. The inner part of the Pd-Ag membrane tube (lumen side) receives the exhaust gas to be treated while hydrogen stream is injected in the shell side in counter-current mode. The Pd-Ag membrane is known to be only permeable to hydrogen and its isotopes (deuterium and tritium), allowing hydrogen to permeate in the lumen side. Tritiated gas HT formed by isotope swamping also permeates to be finally recovered in the shell side of the CMR. Based on previous studies that have allowed defining the most efficient operating conditions for the detritiation process, this work presents the results obtained by the coupling of the detritiation facility with the CMR. Due to safety considerations, restrictions on the nature of the carrier gas were applied, rejecting air as the carrier gas even though air was the best candidate for the detritiation part of the process.

P1-121 **Shutdown dose rate assessment with the Advanced D1S method: development, applications and validation**

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The prediction of the shutdown dose rate induced by neutron activation is a major safety task for fusion reactors, as part of planning the operations of intervention and maintenance in order to guarantee the dose limits can be kept. More than ten years ago ENEA and the ITER team developed the Direct 1- Step method (D1S). It is based on the use of a modified version of the MCNP Monte Carlo code with specially prepared nuclear cross-section data. D1S is one of the most reliable and validated codes for the three-dimensional calculations of the shutdown dose rates in fusion devices. In this approach the decay gammas of the radioactive nuclides are emitted as prompt and thus, the neutrons and decay gammas are transported in a single Monte Carlo particle transport simulation. Time correction factors, calculated with FISPACT activation code, are applied to the scored quantities to take into account the build-up and the decay of the radionuclides considered. The "Advanced-D1S" is an improved version in which new computation capabilities have been introduced, such as the dose rate spatial mesh maps and automated time dependence. The coupled neutron-decay gamma transport allows providing fast results at different times after shutdown in a single MCNP simulation with intrinsic uncertainties in dose rate and decay gamma estimation and identification of contribution from each radionuclide and machine component. The present paper addresses the recent code developments and applications to JET, ITER and DEMO tokamaks. Results of benchmarking with measurements and Rigorous 2-Step (R2S) calculations are summarised and discussed as well as limitations and further development needs. The outcomes confirm the essential role of the Advanced-D1S methodology and the evidence for its complementary use with the R2Smesh approach for the reliable assessment of shutdown dose rates and related statistical uncertainties in present and future fusion devices.

P1-122 **CFD analysis for the transport of tritium within different process rooms of ITER**

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Some buildings from the ITER reactor complex have radioactive material, including tritium. The HVAC provides room air conditioning to these buildings. Within these rooms tritium and radioactive aerosol concentrations in air are monitored. If tritium contamination is detected in air by tritium detectors, the affected room is isolated (inlet and exhaust) and atmosphere detritiation system (DS) is started. A detailed study of transport of tritium within three process rooms is required to evaluate performance of tritium monitoring systems and evaluate spread of contaminants internally. To develop this analysis IDOM has carried out intensive Computational Fluid Dynamics modeling through the use of the commercial software package ANSYS FLUENT© v 13.0. Two different kinds of leakages have been analysed. It has been studied the evolution of the radioactive levels, times to detection for each of the release monitors and the accumulated activity by the exhausting system within each of the rooms, under different scenarios of HVAC regime and temperatures.

Previous to the detailed study of the leak evolution in a number of ITER rooms, a qualification exercise has been carried out by benchmarking a tritium release experiment in the so called Room #01.

P1-124 **Penetration of tritiated water vapor through hydrophobic paints for concrete materials**

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Tritium transfer behavior in hydrophobic paints coated on concrete walls of tritium handling facilities and buildings of fusion reactors are investigated in order to obtain fundamental data of contamination and decontamination of tritium in the case where a large amount of tritium leaks to concrete buildings. Although there are several data on interaction between tritium and concrete materials, there are few data on paints for concrete materials. The present authors focused on epoxy resin as hydrophobic paints, which are usually coated on concrete walls. We measured the amount of tritium trapped in and permeated through the paint films which were exposed to tritium atmosphere which concentration was 650 ~ 700 Bq/cm3. The amount of tritium trapped in epoxy paint films was proportional to thicknesses of the films. Tritium release rate from the films depended on their thicknesses. It was thought that the rate-controlling step of tritium transfer process was not adsorption/desorption on the surface of the paint but diffusion in it. The mass transfer rate of tritium was estimated by an analysis of diffusion model and was also applied to tritium behavior in various paints. The details will be given in the conference. Obtained data will contribute to accumulation of data with tritium contamination and decontamination.

P1-125 **Neutron Spectral Effects on Pb-17li Activation: a Study for different Blanket Designs**

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Research on the eutectic Pb17Li is part of the blanket studies carried out in the US and Europe for ITER and for fusion power reactors. Different blanket concepts using liquid Pb17Li as breeder are being developed: the Helium-Cooled Lithium-Lead (HCLL), the Dual-Cooled Lead-Lithium (DCLL), and the Water-Cooled Lithium Lead (WCLL) blankets. This paper presents the safety aspects of the Pb17Li activation, in particular the neutron spectral effects on the activation products buildup. Particular attention is given to Po-210 and Hg-203, which are recognized as the two most noxious activation products of this breeder. Po210, in particular, is mainly due to neutron activation of Bi, present in the Pb17Li as an impurity. Activation calculations were performed with the EASY – FISPACT code using, as input, the neutron flux throughout the Pb17Li breeder in the three blankets referenced above. In particular, the presence of water in the WCLL blanket makes the neutron spectrum generally softer than the one - for instance - of HCLL, considering blanket zones at the same radial coordinate, i.e., distance from the plasma chamber. The Po-210 and Hg-203 activation cross-sections have a strong dependence upon neutron energy: in particular, the Bi209(n,ã)Bi210 cross section, leading to the formation of Po210 as a decay product, has a strong resonance for high-energy neutrons but lower cross section for epithermal neutrons in the energy rage 1 eV-1 keV. In this paper we present the detailed analysis of the buildup of Po210 and other radionuclides in the Pb17Li breeder. A comparison is made to the inventory of these nuclides in the three blankets under consideration. The spectral effect on the production of the Po210 is emphasized where it is shown that the presence/absence of water is main driving factor in determining the amount of the Po210 generated due to its superior moderation capability.

Topic H FNT

P1-127 **Manufacturing Prototypes for Lipac Beam Dump**

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The LIPAC, formerly known as IFMIF EVEDA accelerator, will have a beam dump that must be able to stop deuteron continuous and pulsed beams with energies up to 9 MeV. The maximum beam power is 1.12 MW corresponding to a beam current of 125 mA. The design of the beam dump is based on a copper cone 2500 mm long, 300 mm aperture diameter, 5-6.5 mm thickness, whose inner surface faces the beam. The cooling is provided by water flowing at high velocity along its outer surface. The manufacturing of the inner cone for the beam dump poses a considerable difficulty due to the slender geometry and the surface requirements. Basically two technical solutions have being considered for the manufacturing of the inner cone made of pure copper, Electron Beam Welding (EBW) and Electroforming. The article will show the learned lessons with the manufacturing of the prototypes as well as other tests performed.

P1-128 **The F4E programme on nuclear data validation and nuclear instrumentation techniques for TBM in ITER**

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Reliable and accurate nuclear data for fusion technology applications are being developed and validated in world-wide collaborations. A particular and coordinated effort has been devoted in Europe since more than two decades both on the evaluation, processing and benchmarking of nuclear data as well as on the experimental data base required for the validation of fusion relevant libraries such as JEFF and EAF. The overall objective is to develop and validate the predictive capabilities, including reliable uncertainty margins, of nuclear analysis tools for utilization in a DEMO design. In recent years, the experimental programme funded by Fusion for Energy strengthened the development efforts on candidate nuclear instrumentation techniques particular for the use in Tritium Blanket Modules (TBM) in ITER. To this end, TBM mock-ups have been irradiated in DT-neutron generators to compare the predictions by Monte Carlo transport calculations with measurements of responses of interest, such as the tritium production rate, from several complimentary detectors.

The paper presents a detailed review on the latest achievements of nuclear data validation activities for ITER, IFMIF and DEMO and on the development of detectors with potential resistance to the harsh environment in TBM in ITER. Among those are activation foils for short term measurements, self-powered neutron detectors, single crystal diamond detectors but also direct tritium monitors employing liquid scintillation techniques. The research objectives of future development, taking into account specificities of TBM design integration and unique testing opportunities in a forthcoming JET DT-campaign, will be described. On-going work in the frame of the FPA-395-01 grant is presented as a continuous and comprehensive support of the European strategy on validated nuclear analysis capabilities for fusion technology applications.

P1-129 **Development on Nuclear Validation Facility and Test Platform for Fusion Reactor in China**

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Fusion blanket is one of the key issues for fusion reactor. Liquid PbLi and solid pebble breeder conceptual designs and corresponding International Thermonuclear Experimental Reactor (ITER) Test Blanket Module (TBM) have been under development in the world, and the related R&D activities are being performed too. However, the calculation and simulation results for their designs need experimental validation, so series nuclear and non-nuclear test platforms such as neutron generator, large coolant loops are under development in the world. High Intensified Neutron Generator (HINEG) is an accelerator-based D-T fusion neutron source, which can produce both intense steady neutrons and nanosecond pulse neutrons and is designed by Institute of Nuclear Energy Safety Technology, CAS. It will be mainly used to perform integral neutronics experiments for fusion reactor. The designed source strength of steady neutrons is 3„e1013 n/s. Its engineering design was completed in 2012, and the facility construction will be finished by the end of 2013.

Forced convection PbLi loop, DRAGON-IV, has been built to meet the out-of-pile test requirement for the Dual-Functional Lead-Lithium TBM (DFLL-TBM) of China. And the corrosion test, magnetic-hydra-dynamics (MHD) effect, flow characteristic and key components test etc. can be implemented in the facility under different test conditions. In addition, the compact DRAGON-V loop will be built in the near future to test the key technology for thermal dynamics etc. of dual coolants for 1/3 scale sized DFLL-TBM before the auxiliary systems DRAGON-VI/VII for EAST-TBM and ITER-DFLL-TBM are constructed. In this paper, the progress of these test facilities will be presented.

P1-130 **Direct Measurements of Particle Flux along Gap Sides in Castellated Plasma Facing Components**

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Plasma deposition into gaps of castellated plasma-facing components is a topic of intense research efforts because of significant consequences in terms of tritium retention issues and power loads for the ITER divertor lifetime. Previous studies of plasma deposition into gaps use 2D kinetic simulations for either predicting the power deposition under ITER conditions [1] or for studying the impurity co-deposition processes involved inside a gap [2]. In this paper, we report results of dedicated experiments made for benchmarking our in-house 2D particle-in-cell (PIC) code that was used for those previous studies [1,2]. A dedicated probe, the so-called Sandwich Probe, has been specially designed for measuring ion saturation profiles along the 2 sides and the bottom of a gap between 2 tiles in poloidal and toroidal orientations. The novelty of such experiments is the real time measurement of the plasma flux inside the gap during a tokamak discharge compared to previous experimental studies which are post-mortem. These experiments were performed in TEXTOR and COMPASS tokamaks and results are compared with 2D and 3D PIC simulations, using the recent 3D upgrade [3] of our in-house code. Due to the finite geometry of the tiles and, in particular, the gap crossing between 4 adjacent tiles [4], the 3D simulations give more accurate results. The plasma deposition in the gap given by 3D calculations matches better the experimental values. References

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P1-131 **Numerical Simulations on Natural Convective Heat Transfer and Active Cooling of IFMIF Test Cell**

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The International Fusion Materials Irradiation Facility (IFMIF) is designated to generate a materials irradiation database for the future fusion reactors. The Test Cell (TC), a cavity with internal dimensions 2.8 x 4.0 x 4.25 (m), accommodates all of the test modules and the lithium target assembly. It will be filled with helium gas at a sub-atmospheric pressure (0.2 bar). Due to the (non-uniform) nuclear heat generation, all the test modules inside the TC will be actively cooled. Other components like supporting structures, pipe lines, cables etc. will be passively cooled by natural convection. The heat will be removed from the steel liners surrounding the TC by active water cooling. This paper concerns the thermo-hydraulic simulations of the Test Cell using Ansys-CFX. The simulation model includes the natural convection inside the TC and several forced convective water flows in the pipelines attaching on the liners. The arrangements of the water pipelines are partially optimized to avoid hot spots on the liners. The numerical models for the natural convection were verified against experiments from literature. The results will be presented in terms of temperature fields and heat transfer performance.

P1-133 **Luminescence Qualification of Radiation Induced Damage and Thermal Recovery in Aluminas**

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In future fusion devices, besides neutrons and gammas, oxide ceramics such as Al2O3 for insulation applications will be subjected to ion irradiation. H and He ions, the predominant components of the residual gas, accelerated by local electric fields can produce an important increase of surface electrical conductivity, which may be as serious as volume degradation in terms of their operating capabilities. This degradation is the result of oxygen loss due to preferential radiolytic sputtering during irradiation in vacuum. The progressive formation of oxygen vacancies (F and F+ centres) and the associated light emission can be used to monitor the material modification, as shown in previous work where a correlation between ion beam induced luminescence (IBIL) and surface electrical conductivity was found.

Limited experimental data indicates that electrical degradation and associated luminescence evolution are slowed down by increasing irradiation temperature. Initial attempts to reverse the degradation by reoxidation produced an apparent full recovery of measured conductivity, but only partial recovery of the luminescence. This may be due to the larger material depth contributing to the IBIL, compared with the surface conductivity, suggesting that luminescence is a more reliable indicator of material recovery.

The aim of this work is twofold: to explore the possibility of post irradiation materials recovery by extended thermal annealing in air, and the collection of sufficient data to permit the use of luminescence as a characterization method for inaccessible components in future fusion devices and present fission reactor experiments.

Surface electrical conductivity and IBIL, as a function of dose and temperature for 3 types of á-alumina and sapphire, are presented before and after different thermal annealing cycles in air. Different characterization techniques have been employed following thermally assisted reoxidation to assess the full extent of recovery as compared with that indicated by the electrical conductivity and luminescence results.

P1-134 **Asymmetry of Wendelstein 7-X magnet system introduced by torus assembly**

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The magnet system of the stellerator Wendelstein 7-X consists of 5 modules of 14 superconducting coils each. After manufacturing the coils and assembly of the modules on temporary stands, the position of each module on the machine base was successfully optimised to minimise the magnetic field asymmetry. The asymmetry originates from inevitable geometric deviations of the coils from the target shape due to manufacturing and assembly tolerances. However, new deviations were introduced after module optimisation due to bolting the modules of the magnet system together to a torus, removal of temporary supports and further loading of the machine base with weight of additional components. In this paper, the geometrical deviations along the centre line of the coil currents are assessed through detailed step-by-step non-linear FE simulation of the assembly procedure of the complete torus. The model is validated against measured displacements and reaction forces monitored during consequent assembly steps. The results are being used to quantify the obtained field asymmetry and countermeasures to minimise it.

P1-135 **Development of a high resolution neutron spectroscopy system using a diamond detector and a remote digital acquisition methodology**

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The need of performing high resolution fast neutron spectroscopy in a very hard environment like that of the Radial Neutron Camera (RNC) of ITER, requires to develop new detectors and methodologies. Diamond detectors have been proved to be excellent candidates but the electronics required to read and process the detector signals needs a paramount improvement. Diamond detectors produce very small and fast output signals which can be easily processed using standard charge amplifiers connected with a short cable to the detector. Because of the high radiation level and the temperatures expected near the detector positions in the RNC, the electronics must be placed several meters away. A novel preamplifier was developed that, connected to a diamond detector using several tens of meters of low capacitance coaxial cable, is able to produce fast output signals suitable to be processed by digitalized electronics. These fast output signals allow to operate at high count rates avoiding pile-up problems. This novel preamplifier connected to a digitizer is here tested in the neutron energy range from 7 MeV to 20.5 MeV using the mono-energetic neutrons produced by the Van de Graaff (VdG) accelerator of the EC-JRC-IRMM and the PTB cyclotron. In both accelerators a deuteron beam was used. It was impinging on a tritiated target during runs in steady state with the VdG, and on a deuterated gas target during the pulsed cyclic operation of the PTB cyclotron. From the measurements the experimental response functions of the diamond detector at different neutron energies were obtained. These response functions have been compared with those predicted with a routine which was implemented for the Monte Carlo code MCNPX with the scope to validate the calculations versus the experimental data. The goal is to develop a tool which allows to calculate the diamond detector response functions in order to perform neutron spectroscopy.

P1-136 **TRIPOLI-4® Monte Carlo code ITER A-lite Neutronic Model Validation**

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3D Monte Carlo transport codes are extensively used in neutronic analysis, especially in radiation protection and shielding analyses for fusion reactor. TRIPOLI® Monte Carlo family has been continuously developed at CEA from the mid 60s. TRIPOLI-4®, the fourth generation, was totally re-written in C and C++ with fuller description of geometrical elements (surface-based and/or combinatorial types) and more precise representation of basic nuclear data (pointwise cross-sections). It is the reference Monte Carlo code for CEA, EDF, and a number of branches of AREVA. The aim of this paper is to show the capability of TRIPOLI-4® to model a large-scale fusion reactor with complex neutron source and geometry. In the past, numerous benchmarks have been conducted with TRIPOLI-4® on fusion applications. Analysis against experiments KANT, OKTAVIAN, FNG and numerical benchmark between TRIPOLI-4® and MCNP5 on the HCLL DEMO2007 and ITER A-lite models (by Y.K. Lee and C. Fausser) have been carried out successively. Nevertheless, in this ITER A-lite benchmark only the neutron wall loading was analysed, its main purpose was to present MCAM (the FDS Team CAD import tool) extension for TRIPOLI-4®. Starting from this previous work a more extended benchmark was performed about the assessment of neutron flux, nuclear heating in the blankets and tritium production rate in the European TBMs (HCLL and HCPB). The methodology to build the TRIPOLI-4® A-lite model is based on MCAM and the MCNP A-lite model (version 4.1). Simplified TBMs (from KIT) were integrated in the equatorial-port. Comparisons of neutron wall loading, flux and nuclear heating show a good agreement between the two codes. Discrepancies are mainly included in the Monte Carlo codes statistical error.

P1-137 **Modeling of Hydrogen Isotope Retention in the Tungsten Divertor of EAST under different discharge operations**

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Tungsten is foreseen as plasma-facing materials (PFM) in modern fusion devices due to its beneficial properties. Under normal conditions, hydrogen isotope (HI) retention in W is not a severe issue, but it can be greatly raised in the presence of damage, created by neutrals and ions of high energy during the tokamak discharge. Therefore, the issue of HI retention inside the W material is still a concern. It was observed experimentally that the energy and fluence of incident particles on the divertor plates, which have big effects on the HI retention [1], depend on the discharge operation.

In this work, a suite of codes, SOLPS-SDTrimSP-HIIPC, has been applied to study the HI retention in the W divertor of EAST, whose PFM will have been updated to W soon. In this suite, the HIIPC code [2] is used to model HI retention in the W divertor plate with the incident particle and heat fluxes obtained from the SOLPS code [3], and the implantation depth evaluated from the SDTrimSP code [4] as the input parameters. The HI retention after repeated pulse discharge is simulated and the effects of the pusle duration and frequency are discussed. It is shown that the pulse duration increases the retention amount significiently; however, the time between two pulses decreases it. During the H-mode operation, large particle energy and flux in the form of ELMs would deposit onto the divertor target. The HI retention under this condition is also studied and compared with the L-mode discharge operation. The simulation results reveal that the ELMs can increase the amount of HI retention due to the deeper implantation depth and smaller HI release rate on the plasma facing surface.

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P1-138 **Radiation and momentum exchange in the divertor detachment induced by gas puffing: PIC-DSMC simulation**

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Divertor plasma detachment will be chosen as the operation regime for ITER to satisfy the requirements, mitigating the heat flux to the divertor wall and removing helium ash and other impurities. The divertor detachment involves complicated chemical-physical processes, which include interactions between charged and charged particles, charged particles and neutrals, and neutrals and neutrals. Particle-in-cell (PIC) method can cope well with charged particles but not with neutrals [1, 2], which can be simulated with the direct simulation Monte Carlo (DSMC) method [3]. In this work, a parallel code based on the self-consistent PIC-DSMC model has been developed to study the divertor detachment induced by gas puffing, which is one of the most important methods to facilitate the divertor plasma to obtain the detachment to the divertor targets. Apart from charged particles, the states of neutrals of different excited states (D(n=1,2,3,4,5 and Ar(1,2)) and D2 are tracked to deeply understand the contribution of the impurity radiation to heat exhaust in the divertor region. To evaluate the relative intensity of Balmer-series line radiation, a radiation model like the collision-radiation model [4] is employed. The relative intensity from the simulation can be compared directly with the corresponding experimental results. In the simulated scenarios, the Ar and D2 gases are pumped into the divertor region, respectively, and the corresponding detachment processes are studied. Physical quantities such as the momentum loss, particles and energy fluxes to the divertor target, the temperature and density distribution of every type of tracked particles are monitored. From the kinetic simulation, the momentum-loss distributions of plasma flux resulting from recombination and charge exchange are given, respectively, and relative contributions from recombination and charge exchange to the plasma detachment are quantitatively analyzed.

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P1-139 **Development of ITER IOIS Assembly Tool and Mock-up**

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The ITER toroidal field coils (TFCs) are connected by 3 different types of connecting structure as follows; Outer Intercoil Structure (OIS), Inner Intercoil Structure (IIS), Intermediate Outer Intercoil Structure (IOIS). One of them, the IOIS is composed of a front and rear plates and 6 customized pin connections located at the upper and lower areas of the TFC. The IOIS are assembled in the assembly building between two TFCs of one sector during sector sub-assembly phase and in in-pit between neighboring sectors during sector assembly procedure. According to IO¡¯s modified IOIS assembly procedure, the IOIS front and rear plate should be handled and installed by the dedicated assembly tool considering very limited man access and crane accessibility conditions. When the rear plate of the IOIS is inserted there is a very limited space between the vacuum vessel thermal shield (VVTS) and the IOIS wings. Also, the front and rear plates of the IOIS should be controlled and assembled simultaneously. The two plates of the IOIS should be positioned and supported temporarily at their installation locations before arbor operation and pin connections. The design of the dedicated assembly tool for the IOIS assembly has been developed to use both during sub-assembly and final assembly phase by the Korean domestic agency (KODA). Adjustment system of this assembly tool, electrical handling system for control of the TFC front and rear plates has been also designed to meet the functional requirements requested by IO. For design verification of the IOIS assembly tool mentioned above, a mock-up has been fabricated in half size and tested according to the IOIS assembly procedure. Some problems of current design and improvements will be presented also.

P1-140 **Production and validation of a 3D-printed coil frame for the UST_2 modular stellarator**

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Geometric complexity of stellarators hampers a straightforward production of conceived optimised magnetic configurations. Integration of the engineering design with new fabrication methods may reduce the production cost and accelerate the production process. A fast cycle production of experimental fusion devices also might result in a faster advance in fusion plasma science. Several different stellarators could be used to test configurations for improved turbulent transport or to validate new divertor configurations. In this framework, and based on the results from the previously built UST_1 stellarator, the present work try to study and validate the feasibility of 3D printing methods (additive manufacturing) for small experimental stellarators. The paper summarizes the engineering development, fabrication and validation of a coil frame test sector for the UST_2 stellarator. The definition of the Last Closed Flux Surface and winding surface for the test sector is based on an optimised quasi-isodynamic poloidal stellarator. A Hollow Sparse coil frame concept is developed to still keep low the cost in spite of the present expensive 3D printing materials and printers. A winding configuration is developed to allow fast and accurate winding of the conductor in the modular coil frame grooves.

Topic I Repair and Maintenance

P1-141 **Concept design on RH maintenance of CFETR Tokamak Reactor**

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CFETR which stands for Chinese Fusion Engineering Testing Reactor is a superconducting tokamak device. The concept design on RH maintenance of CFETR has been done in the past year. As we all know, the RH maintenance is one of the most important works for Tokamak reactor. The fusion power was designed as 50 - 200 MW and its duty cycle time (or burning time) was estimated as 30-50%. The center magnetic field strength on the TF magnet is 5.0 T, the maximum capacity of the volt seconds provided by center solenoid winding will be about 160 VS. The plasma current will be 10MA and its major radius and minor radius is 5.7 m and 1.6 m respectively. All the components of CFETR which provide their basic functions must be maintained and inspected during the reactor lifetime. Thus, the Remote Handling (RH) maintenance system should be a key component, which must analysis as detailed as possible during the design processing of CFETR, for the operation of reactor. The main design work for RH maintenance in this paper was carried out including the position and size of maintenance ports, demolition of the damaged parts and structure design of RH. In addition, the RH structure design and optimization based on different types of maintenance ports were detailed discussed in this paper. What is more, the technical problems encountered in the design process will also be discussed as clearly as possible.

P1-142 **Determination of capsule position by monitoring flow-rate in ITER neutron activation system**

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ITER Neutron Activation System (NAS) measures the neutron fluence to the first wall, providing the total fusion energy evaluation from the plasma during the irradiation period. The encapsulated activation sample is transferred pneumatically via the metallic tube assembly between the counting station and the irradiation station. The reliable monitoring method for the location of the sample is essential to ensure the diagnostic accuracy of the NAS and to recognize accidents such as a clogging of the sample. However, conventional methods using mechanical, optical or electrical detectors to determine the capsule position cannot be used in the ITER NAS because of their limited space as well as extremely high magnetic and radiation fields. In this study, a new method for the reliable determination of the sample position using the flow-rate change inside the tube assembly is presented. Experimental results confirm that the temporal variation of flow-rates at both inlet and outlet positions of the transfer tube assembly provide information for the location of the sample with high accuracy. This method is expected to improve the accuracy, operational stability and the ability to handle an accident such as the capsule stuck in ITER NAS.

P1-143 **A three-layered model for generic description of remote handling maintenance tasks in supervisory control systems**

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Complexity and diversity of ITER maintenance tasks require the use of a large set of remote handling equipment supervised from a control room by human operators organised in work cells. Due to restricted visibility conditions, such tasks require the use of advanced Supervisory Control Systems able to provide operators with efficient virtual representation of the environment and context-based assistances so as to optimize operations effectiveness, reliability and safety. As it would be highly inefficient to develop specific supervisory control applications for each task or equipment, this work, carried out in the EFDA GOT RH project, aims at providing a language and tools to define in a generic way ITER maintenance tasks and scenarios for in-vessel and hot cell operations. A three-layered approach is proposed to fulfil this objective: interfaces, behaviours and interaction modes. Interfaces define actions opportunities on the scene elements. Behaviours define the high-level functions supported by the interfaces and interaction modes define the way behaviours are achieved in terms of task allocation between human and machine.

This model is integrated in an architecture taking into account prescribed procedure for maintenance tasks and including tools for planning and task allocation. The proposed approach aims at modelling interactions between operators and supervisor in the control room in order to define shared actions plans between work cells. Each task of the procedure can be defined with a given interaction mode which can be dynamically adjusted according to the situation. The proposed formalism intends to be linked with intent recognition and procedure monitoring in order to propose more relevant context-based assistances to the operator. Depending on the interfaces defined on the objects and the task to be achieved, human-machine interfaces can thus be dynamically adapted to indicate the next action to perform and more generally speaking, to enable the required supervision functionalities.

P1-144 **Implementation of Multibody analysis in the Verification and Validation process of ITER Remote Handling Systems**

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The work behind this paper takes place in the EFDA's European Goal Oriented Training programme on Remote Handling (RH) "GOT-RH". It is written under the project WP1.5: "Verification and Validation (V&V) of ITER RH system requirements using Digital Mock-Ups (DMU)". The purpose of this project is to study and develop efficient approach of using DMU's in the V&V process of ITER RH systems.

Complex engineering systems such as ITER facilities lead to substantial rise of cost while manufacturing the full-scale prototype. In the V&V process for ITER RH equipment, physical tests are a requirement to ensure the compliance of the system according to the required operation. Therefore it is essential to virtually verify the developed system before starting the prototype manufacturing phase. One important issues for the design of ITER divertor maintenance system is to take into consideration the flexibility of the systems under loading conditions.

The paper describes the implementation of multibody simulation activities in the V&V process of ITER RH systems. According to the verification roadmap, multibody analysis is the first stage of the verification process. The paper focus on the static aspects of the multibody simulation, which consists of combining finite element analysis, joint tolerance analysis and assembly design in the same multi-modal model. As a result we were able to visualize a more accurate representation of the real behaviour of the developed system. Consequently, simulations results are displayed, and comparisons between the virtual and the real system are performed. We discuss further the expected results and futures activities.

P1-145 **Interactive Virtual Mock-ups for Remote Handling Compatibility Assessment of Heavy Components**

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One of the challenges to make ITER to a success is it maintainability by Remote Handling (RH). For RH class one –and critical class two- components it is obligatory to build hard ware mock-ups to validate their RH compatibility. But full scale mock-ups of large ITER components -that can weigh up to 45 tons- are very expensive in time and cost. A promising alternative approach is the use of an interactive virtual reality simulation with real time rigid body dynamics and contact interaction.

This paper explores the used of an interactive virtual mock-up for early stage RH compatibility analysis of heavy ITER components. The mock-up is based upon CATIA models. These models are adjusted in 3DS MAX for real time rendering and physical behaviour. Specific material and contact dynamics characteristics are added and finally the models are simulated with the Virtual Slave Simulator [1]. Connected to the simulator are a keyboard to control camera and crane movements and a Haption Virtuose™ 6D to control a dextrous manipulator.

The simulations results are used to estimate the required maintenance time, identify bottlenecks and propose design improvements. Further the paper discusses various ways to validate the rigid body dynamics and the limitations on use of virtual mock-ups. Finally it proposes improvements for more realistic and reliable simulation behaviour.

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P1-146 **TAO3: modular controller software for highly interoperable force feedback teleoperation**

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Within ITER project as well as in other remote handling (RH) application fields, operations to be remotely performed will be diverse in terms of load, accuracy, workspace, etc. Consequently, ITER remote maintenance systems (IRMS) will include a large number of different slave systems. Nevertheless, they will be controlled from a reduced number of standard and centralized control stations possibly using only one kind of standard force feedback master arm. Coupling technologically and cinematically heterogeneous master and slave systems has been made possible by computer-aided teleoperation systems such CEA LIST's TAO2000 software platform. TAO real time controller was designed to be generic but, although designed as a whole entity, deployment is always done on two physical master and slave controllers communicating through low latency channel in order to guarantee adequate haptic rendering and operated from high-level control, tuning and monitoring client applications. The basic idea we want to highlight is the following: why not to (re)distribute controller's functionalities over the system according to present task needs, available resources, experienced problems, etc? Under this scheme, a functional component could interchangeably run on either one or the other physical controllers or even on a third party computer.

In order to demonstrate this concept, we started redesigning TAO using Orocos framework. After justifying the choice of Orocos as middleware, we present three aspects of our new TAO3 software architecture: (i) system functional analysis; (ii) Orocos component-based implementation; (iii) deployment including first experimental results. In addition, we also discuss TAO3's compliance with respect to ITER requirements.

This contribution reports works carried out by CEA LIST under EFDA GOT RH program (WP1.4: IRMS interoperability) and ISO standardization activity on remote handling for nuclear applications (ISO-TC85-SC2-WG24).

P1-147 **Assessment of a Rate-Position Controller for Remote Handling in Nuclear Fusion Maintenance Tasks**

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This work assess a novel haptic rate-position controller. This controller can be applied for remote handling a slave robot in a large workspace by means of a small haptic master device. It controller has demonstrated its efficiency when the master device is significantly smaller than the slave device. While performing the task, the controller displays haptic information to the operator in order to inform when a change in the operation mode occurs and when the slave robot interacts with the environment. This hybrid approach for a remote robot guidance differs form the classic position guidance but several works have studied the advantages and disadvantages of this two control modes, position and rate, in a individual sense.

In order to verify the feasibility of the proposed approach, a virtual slave robot performs simulated tasks such as a IFMIF (International Fusion Material Irradiation Facility) remote handling task consisting in the manipulation of test materials irradiated in a nuclear environment. In this task the remote handling system inserts or removes a sample, called rig, into/from the high flux test module (HFTM) where the irradiation occurs. The operator controls the slave robot using a commercial haptic device, called Phantom Omni, with 6 DoF, three of them actuated allowing to display force vectors as feedback.

The proposed algorithm has been compared with classic architectures as position-position, force-position and four channels. Results obtained show an improvement in the accuracy using the proposed method and a similar task execution time in comparison with the other architectures. The main strenghts of this method are: the possibility to control a slave with large workspace using a haptic desktop master device and great accuracy achieved due to the two operation modes.

P1-148 **Griping tool for the ITER upper port plug RH extraction/insertion**

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The conceptual design of several gripping tools and their mechanical interfaces are being carried out for the ECH UPP for ITER within the WP10-GOTRH programme. EFDA finances the GOT RH (Goal Oriented Training Programme for Remote Handling) with the objective of training engineers in remote handling and in the fusion field. The purpose of this paper is to introduce new concepts of gripping tools for the plug extraction/insertion in the upper port of ITER. All these gripping tools are designed according to IO input data and geometrical constraints. The gripping tools have to be able to extract/insert the plug in the case of maximum misalignment between the plug and the tractor without clashing with other components. The paper also defines the functional requirements the gripping tools need to comply with; such as modularity of the assembly, avoidance of force feedback in the RH devices, recovery/rescue of the gripping tool, requirements related to the friction to avoid the jamming of the plug skids with the port rails, misalignment to be absorbed by the gripping tool in the plug insertion/extraction, etc. The requirements and input data are verified and validated through 3D simulation with Catia mock-ups of the gripping tools. The strengths and weaknesses of each gripping tool model are compared in order to propose one of them to be developed for the plug extraction/insertion in the upper port of ITER within the GOT RH programme.

P1-149 **Software fault detection and recovery in critical real-time systems: an empirical study**

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Context: Remote handling systems are used to inspect, make changes to, and maintain components in the ITER reactor, being an example of mission critical system. Failure in a critical system may cause damage and significant financial losses, making dependability one of their most important properties. However, even if a software system has been developed using valid development process, the system might still fail due to undetected faults, hardware failures etc. Critical systems therefore need to be able to resume operation after faults have occurred, but design of effective fault detection and recovery mechanisms poses a challenge due to timeliness requirements combined with growth in scale and complex interactions in computer systems.

Objective: Several frameworks and programming languages support use of decoupled architectural models that can be used to implement and support fault tolerance solutions, but this kind of approach is typically used in server-based applications. Our goal is to evaluate effectiveness of this approach when used in a mission-critical remote handling (RH) control system.

Method: We use an experimental RH system to control an industrial manipulator in ITER relevant RH task scenarios. Architecture of the RH control system prototype is based on the service-orientation paradigm, adapted with data-centric design principles to better suit the needs of distributed real-time control systems. The decoupled architectural model is used to implement safe management of services (processes) in the RH control system to detect failures based on observing abnormal behaviour.

Results: The decoupled architecture model supports handling of transient faults and implementing additional design patterns for fault tolerance even in critical real-time systems like remote handling control systems. The recovery process may slightly reduce availability of some services and provides no recovery for persistent faults unlike redundant fault tolerance solutions, but improves system robustness and requires less additional code, thus providing a cost-efficient solution.

P1-150 **Human-in-the-loop tele-operated maintenance: what can ITER learn from JET?**

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Remote maintenance is a key issue for sustainable operation of future fusion plants such as ITER. At JET (the first and only fusion device with a fully operational Remote Handling (RH) system), the maintenance team has already built up an extensive body of experience with complex real-world RH. A human-in-the-loop approach was found to be crucial due to the essentially unpredictable nature of the maintenance required. The flexibility of the human approach is however counterbalanced by the extensive operator training, a high operator workload, and long execution times.

The aim of this paper is to analyse the body of experience from recent shutdowns at JET, investigating the main limitations and challenges of the current human-in-the-loop RH operations at JET and identify the key areas for further improvement. The analysis is based on a series of interviews with experienced master-slave operators and RH responsible officers as well as detailed investigation of the task logbooks. Although the master-slave tasks itself are the core of the Remote Handling, a significant part of the operational time and effort is spent on additional assistive processes such as slave positioning, getting the correct visual feedback and the logistics of tools and components. The operator interviews reveal that the availability

of appropriate visual feedback has a significant influence on the time to complete operations. Also the type of RH tasks and the configuration of master with respect to slave has a major impact on performance. These findings are supported by objective data which is routinely and meticulously logged during all RH interventions performed at JET. The results found in this study will be used for future research and development activities, with the aim to further design and optimize RH maintenance systems for ITER and beyond. A promising direction of research is the applicability of haptic guiding strategies.

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P1-151 **Reliability Requirements Management – addressing Remote Handling controller reliability via probabilistic methods**

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Reliability assessment and verification can be a tedious and expensive problem for system suppliers, as reliability data on component basis may or may not be available and, depending on system size and reliability requirement, empiric testing can consume copious amounts of time and money. This includes both the mechanical structure of the device and the equipment controller system.

The ITER organisation has presented exact reliability and availability requirements for both maintenance equipment and their controllers. Addressing the equipment reliability has been demonstrated in a prior paper; this paper extends the ideas presented in device domain to software. While it must be admitted that technically software failures are deterministic in nature, they can appear as purely stochastic phenomenon for the end user. Following this line of thought, the probabilistic approach was selected. Often in the RAMI process, software is overlooked – not least because the amount of different failures software can encounter and cause. The presented probabilistic method accounts for software failures from the design phase onwards. The results of the allocation can be used as requirements for subcontractors, both component and software. Furthermore, the results can indicate a fault in system design, requiring too reliable components or software to meet the enveloping requirements, possibly necessitating a system redesign. In this paper, a previously presented allocation for a water hydraulic manipulator is augmented to include the controller allocations, slightly altering the original allocations to include the controller in the ITER specified reliability and availability requirements. Maintenance time allocations and downtime analyses – while they could have been performed – were omitted, partially due to the deterministic nature of software failures, which complicates assessing software reliability after a failure. Regardless, if reliability requirements including both software and hardware need to be analysed, the presented method can offer a powerful tool, in both design and verification.

P1-152 **ITER EC H&CD Upper Launcher: Design Options and Remote Handling Issues of the Waveguide Assembly**

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The ITER Electron Cyclotron Heating and Current Drive Upper Launcher (EC H&CD UL), developed by the ECHUL-CA Consortium of Euratom Associations (CRPP, CNR, ITER-NL, IPF, IPP, KIT and Politecnico di Milano), is presently in its final design phase. The study presented here deals with design options and Remote Handling issues related to the waveguide assembly (WGA), an ensemble of mm-wave transmission line components mounted on a vacuum flange. This assembly is part of the primary vacuum boundary of the ITER vessel. This paper describes the preliminary assessment of the RH compatibility of the sub-assembly and a conceptual description of the maintenance actions to be performed on it. A trade-off study has been performed, comparing two possible configurations for the tapers: a Waveguide Integrated Design, in which tapers are integral part of the in-plug waveguide, and an Auxiliary Shield Integrated Design, in which the tapers are integrated into the Auxiliary Shield. An important aspect in the design is to ensure the Remote Handling compatibility. Due to lack of space, the general approach in defining the maintenance strategy is to avoid breaking the interfaces of the different components of the WGA, and therefore to extract it from the Upper Port Plug as a single assembly. The description of the proposed replacement procedure and the required tools are presented here.

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P1-153 **Evaluation of a reconfigurable Modular robot system for inspection and maintenance tasks in nuclear fusion facility**

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This work evaluates a Modular robotic system to be used at large accelerator facilities (e.g. CERN) to reduce human radiation dose during maintenance. We asses the use of the same modular robot system in nuclear fusion facilities for inspection and maintenance tasks, as there is a strong correlation in the environment and requirements at both the facilities. The need to have a robust and safe system to operate in the ionizing radiation facility and reduce human interaction with the hazardous environment. Reconfigurable Modular robot systems (MRS) present advantages in comparison to conventional robots, due to the modular structure which leads the ability of the MRS to achieve various morphologies and therefore achieving different robot structures for task execution. The reconfigurability ability of MRS enables the robot to execute task with optimal solution. The modularity and the reconfigurability in the robot also leads to reconfiguring the robot to recover from faults and the flexibility to change, modify or replace modules depending on the task. The capabilities of the MRS can continue to evolve depending on the requirements without the need to start development of different conventional robot platforms.

The proposed MRS is evaluated through simulations and a prototype across selected morphologies of the robot configuration to perform tasks of remote inspection, radiation survey and remote execution of maintenance task using the various modules of the MRS. Results obtained show the competence of the MRS to perform the tasks as well as its ability to adapt and evolve depending on conditions. The test case shows the use of the MRS as an robot arm to perform teleoperation tasks in the workspace, also as a mobile robot platform to reach the workspace and a combination of both.

P1-154 **Progress in the design of the ITER Neutral Beam Cell Remote Handling System**

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Among the different heating systems foreseen in ITER, two neutral beam injectors delivering each 16.5 MW will be hosted in the Neutral Beam Cell, supported by a diagnostic neutral beam system (the addition of a third heating injector is not excluded at present). The NB cell also hosts plasma diagnostic systems situated in 4 upper ports plugs, together with an additional neutron flux monitor and an in-service inspection system for the monitoring of the vacuum vessel, all accessible via a mezzanine floor. The ITER NB Cell will present a range of technically challenging maintenance tasks whose successful completion by Remote Handling means will be crucial to ensuring operational availability of the ITER machine. Following completion of the conceptual design of the NB Cell RH system, the European Domestic Agency is preparing its in-kind procurement. This paper will focus on recent achievements obtained in view of the next design and manufacturing stages. In particular, central to the envisaged equipment is the nuclear-grade crane, used for lifting and transporting plant equipment and tools within the NB cell, also enabling transfer to and from the ITER Hot Cell. Due to NB cell geometrical constraints, a Monorail Crane, travelling on a monorail track with branches and associated rail switching mechanisms to reach the various work areas, has been designed. The cell layout excludes manipulation of a power and signalling umbilical to the crane trolley. Operating in a nuclear environment will also require all on-board equipment to be adequately radiation tolerant. Consequently, a conceptual solution has been outlined, featuring a conductor bar to supply power to the Monorail Crane, wireless communications and radiation tolerant on-board components.

This paper describes developed concepts for the Monorail Crane with an embarked control and communications system, and complementary designs of specific tooling for NB cell maintenance.

P1-155 **Design of a MGy radiation tolerant resolver-to-digital convertor IC for remotely operated maintenance in harsh environments**

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3. ITER Organization, Saint Paul Lez Durance, France

During the lifetime of ITER, remotely operated equipment is required to support initial installation, planned maintenance and repair of neutron-activated components, and its decommissioning. A key challenge to be met by the European domestic agency to fulfill its in-kind procurement obligations is to ensure remote handling systems can operate under ITER relevant conditions, including high radiation levels up to 1 MGy, elevated temperatures, etc. This requires robust sensors and actuators, some of which also need embarked electronics, such as the read-out of position sensors. This paper presents the design of the first MGy radiation tolerant integrated 16 bit Resolver-to-Digital Converter. RDC's are used to readout resolvers, the most common industrial angular position sensors. A resolver usually consists of a primary rotor winding and two orthogonal secondary stator windings, which produce output signals via magnetic induction. The RDC is custom-designed for commercially available resolvers identified for divertor maintenance operations, which should withstand a radiation dose up to 1 MGy. However, the IC could serve various other applications requiring a guaranteed functionality in extreme environments. The presented RDC uses two integrated ΣΔ Analog-to-Digital Converters (ADC's) to digitize the two resolver outputs. Each ADC features 16 bit resolution for a bandwidth of 10 kHz and utilizes a correlated double sampling technique to remove any radiation induced offset and 1/f noise. The front-end circuit features a static angular resolution of 16 bits (20 arcsec) and a resolution of 10 bits (20 arcmin) at a rotor speed of 100 rps. The circuit relies on a mainstream 130 nm CMOS technology and has a simulated radiation tolerance exceeding 1 MGy. The paper will also demonstrate its ability to operate under temperatures up to 125 °C, and to allow multiplexing with digitized signals from other conventional sensors, extremely useful for embedded redundancy in compact and robust read-out architectures.

Topic L Fission- Fussion

P1-156 **MCNPX/ANSYS Fluent automatic coupling software**

Fabbri, Marco; Colomer, Clara; Alemán, Agustín; Salellas, Jordi

IDOM Group, Barcelona, Spain

Determining the volumetrical power deposition generated by radiation sources through a Monte Carlo code and imposing it into a CFD code, with high accuracy, is a great challenge and can substantially improve the design of any fission and fusion component. The IDOM team has successfully coupled MCNPX and ANSYS Fluent, which are two reference codes for ITER, building an automatic interface software written mostly in C/C++. Basically, once the volumetrical power heat deposition is estimated by a MCNPX mesh tally, the values are mapped with the Fluent mesh and inserted as source term in the User Defined Memory through the User Defined Function or UDF. The mapping process is performed by the MapLib numerical library of the Fraunhofer-Institut which is incorporated into the UDF. If the CFD temperature analysis shows a significant temperature difference from the MCNPX model, a new Monte Carlo calculation will be done varying the cross sections, the temperature and the density of the materials in order to adjust the source terms. An iterative procedure between the codes can be done without any problem. The analysis of several verification and validation cases, as for instance the ITER Triangular Support, has allowed the validation of the procedures and a series of internal checks evaluate the results/error in each step. The interface boundaries and the geometric differences between the two models are the main error sources despite the capacity of the software to deal with all the MCNPX mesh tally typologies (i.e., rectangular, cylindrical and spherical). Anyway, thanks to a deep parameter sensitiveness, which has covered the mapping algorithms, the mesh sensitivity, the typology of element and its connectivity, the whole power deposition error in the process is less than 0.5%.

P1-157 **Conceptual Study of Fusion-Driven System for Nuclear Waste Transmutation**

Hong, Bong Guen

Chonbuk National University, Jeonju, Korea

Conceptual study of a fusion-driven system with a LAR (Low Aspect Ratio) tokamak as a neutron source is performed. An optimal configuration of a LAR tokamak neutron source with respect to both a transmutation rate and a tritium breeding ratio, for an aspect ratio, A in the range of 1.5 to 2.0 is found. Characteristics of both transuranic actinides and minor actinides transmutation is investigated and their characteristics are compared. When transuranic actinides are loaded in the blanket, neutron multiplication factor decreases from the initial value, keff = 0.95, but with the minor actinides loaded in the blanket, the neutron multiplication factor shows a peak value during burn-up and the peak value can be controlled to be 0.95 by adjusting the blanket dimensions. It is shown that a transmutation reactor with a LAR tokamak neutron source producing fusion power less than 150 MW can transmutate the transuranic actinides contained in the spent fuels produced from more than two PWRs (1 GWe capacity) with a production of the fission power greater than 2 GW. Equilibrium fuel cycle analysis is performed and a result is discussed.

P1-158 **Rigorous 2-step shutdown dose rate calculation method based on mesh tally and its application to CLEAR-I**

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Operation of nuclear devices requires a reliable maintenance scenario, one of the major issues is the wait-time after shutdown of the devices until personal access to the devices can be allowed for maintenance or repair, it is very important to estimate the dose rate distribution after shutdown. Since the cell tally based dose rate calculation code has many disadvantages. The shutdown dose rate calculation code based on mesh tally was developed for nuclear devices with large dimension and complex geometries.

The code coupled the Monte Carlo particle transport calculation code with the activation simulation code FISPACT. It can extract the energy and spatially dependent neutron flux distribution from mesh tallies of transport calculation, calculate the homogenized material composition in each mesh and generate input files for activation calculation automatically. It also can analyze the activation results and written source files for decay photons. Besides, it integrated a source subroutine to sample the particles described in source files for transport calculation of decay photons. The dose rate calculation code supported both Cartesian coordinate and cylindrical-coordinate mesh tallies, making it more flexible and applicative than the traditional rigorous two-step method based on cell tally. The correctness of the code has been tested by the FNG experimental benchmark which is designed to validate the shutdown dose rate calculations for the International Thermonuclear Experimental Reactor (ITER). The agreement of the Calculation results and the experiment values was fairly good. The code was then applied to calculate the shutdown dose rate of the China Lead-Alloy Cooled Research Reactor (CLEAR-I). According to the results, the dose rate in the reactor plant was less than 1 uSv/h after shutdown for 7 days, allowing hands-on maintenance. The verification and validation demonstrated the suitability and accuracy of the code for handling shutdown dose rate analyses of large and complex nuclear devices.

P1-159 **Integral neutron experiments for fusion -fission hybrid energy reactor**

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We propose a preliminary design for a fusion- fission hybrid energy reactor (FFHER), based on current fusion science and technology and well-developed fission technology. We use uranium alloy as fuel and water as coolant. The uranium fuel can be natural uranium, LWR spent fuel or depleted uranium. In order to check the conceptual design of sub-critical blanket, an alternate depleted uranium/polyethylene-shell simulation device was established, to carry out uranium-238 neutron capture rate experiment using activation technique. The depleted uranium foils were activated at 90◦ to the incident. By measuring the 277.6 keV Gamma-ray emitted from 239Np generated by 238U(n,Gamma) 239U reaction and correcting self-absorption of uranium experimentally, the 238U(n,Gamma) reaction rate and a total neutron capture of the system were obtained. The experiment was simulated using MCNP code with ENDF/BVII libraries. The simulations and measurements accord within 5% for the 238U(n,Gamma) reaction rate and within 1% for the total neutron capture rate of 238U.

P1-160 **The Source Neutrons and Fuel Distribution Importance for Power Generation and Heat Transfer In Fusion-Fission Hybrids**

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Nuclear fusion-fission hybrid reactor in the form of Mirror system, distinguished by its small size, thus – costs has also some more sophisticated positives. One of such advantages of fusion-driven systems is the neutron yield from fission induced by D-T neutrons, much higher than the one due to fission neutrons. Such effect improves the neutron balance in the system without heightening of its keff, i.e. without impairment of its safety. Besides, the source neutron generation, thus the neutron flux distribution is strongly uneven in axial direction and different in the radial and axial directions of the device. As a result the fuel composition affects the power (heating) distribution. In this paper we raise the issue of fuel composition and distribution in order to reconcile rather inconsistent objectives. On one hand, for sake of safety one should maintain possibly high values of k-source (i.e. expose fissile materials to 14MeV neutrons) and on the other – effectively incinerate MAs (i.e. just the latter ones expose to fast neutrons). Moreover, for optimum cooling the heating distribution (power density distribution) should be kept as flat as possible. In this view, significant is also the heat transfer from the plasma to the coolant through the first wall. This item is of special concern due to the importance of mechanical properties of structural materials of the device. All the calculations were performed using MCNP5 for neutron transport and energy production and multiphysics ANSYS Workbench code for heat transfer simulation. The obtained results of calculations indicated satisfactorily uniform heat generation, owed to appropriate fuel distribution. However, for low plasma Q, due to heavy heat load of the surfaces facing plasma, the problem of its cooling proved harder than expected.

POSTER SESSION 2
Wednesday 18th September
11:00 – 13:00
Ground floor

Topic A First Wall

P2-002 **Numerical Solutions for Liquid Metal MHD Flow in an L-Bend under a Uniform Magnetic Field**

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MHD plays a great role in R&D of a liquid metal fusion reactor blanket. In order to simulate the MHD effect of liquid metal fusion blanket, a FVM CFD code is developed from OpenFOAM toolbox by employing a consistent and conservation scheme for calculation of the Lorentz force. The solver is firstly validated by comparing with analytical solutions on a rectangular duct and circular pipe at a high Hartmann number, and then the solver is applied for simulation of a L-bend duct relevant to fusion blanket. The results show that the flow region is divided into cores and boundary layers of different types in both radial duct and toroidal duct. In the toroidal duct which is short and perpendicular to the magnetic field, a high velocity jet occurs in the side layers and became smaller in the corner of the L-bend. The volume flux carried by the side layers in toroidal duct is more than in the poloidal duct. There is two parts split from flux in side layers of toroidal, one of them is carried by the side layers of poloidal duct and the other part is carried by the core flow of poloidal duct. The Hartmann layer is near the wall perpendicular to the magnetic field, the thickness of this layer is O (Ha-1). In this very thin layer, flow velocity drops sharply from the core velocity to zero (no-slip boundary condition), and the Hartmann layer disappears in the poloidal duct which is parallel to the magnetic field. The flux carried in Hartmann layer enters into the core and mixes together. The simulation shows that the flow distribution change sharply from the toroidal duct to poloidal duct, the flux carried in both Hartmann layers and side layers trend to enter the core part in the poloidal duct and mingle together. This effect may promote the heat transfer from the plasma facing first wall to liquid metal flowed in the channel.

P2-003 **High heat flux testing of Normal Heat Flux First Wall (NHF FW) Mock-ups with calibrated defects**

Bellin, Boris[1]; Banetta, Stefano[1]; Zacchia, Francesco[1]; Davydov, Vladimir[2]; Kuznetsov, Vladimir[2]; Rulev, Roman[2]; Eaton, Russell[3]; Mitteau, Raphael[3]; Raffray, Rene[3]

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The ITER NHF FW panels, able to withstand heat flux up to 2 MW/m2, are currently on-going the final design review process and qualification activities in order to release the manufacturing of the 218 panels to be delivered by F4E.

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

P2-004 **Study of laser-removal and structural changes of W:Al:C layer with Deuterium content**

Kubkowska, Monika[1]; Kowalska-Strzeciwilk, Ewa[1]; Gasior, Pawel[1]; Skrzeczanowski, Wojciech[2]; Fortuna-Zalesna, Elzbieta[3]; Grzonka, Justyna[3]

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Paper presents experimental results of laser-removal of mixed material layers of tungsten, aluminum and carbon with deuterium content on tungsten substrate. Experiments were performed in a vacuum chamber under the initial pressure equal to 5x10^-5 mbar. A Mechelle5000 optical spectrometer equipped with a iCCD (iStar) detector was used to record spectra emitted from the plasma plume. As a source for removal a Nd:YAG laser with fluence about 25 J/cm2 was used. Basing on the recorded spectra it was possible to estimate plasma parameters (electron temperature and density) during the process of laser removal. It was observed that for 25 J/cm2 the mixed materials W:Al:C layer of about 3 microns thickness has been removed after 10 laser shots while Deuterium layer (D-alpha 656.2 nm) (3%) was removed after 1-2 laser shots. Presented study was aimed on: (i) the investigation on the temporal evolution of the individual spectral lines of sample elements and (ii) the optimization of the LIBS (Laser Induced Breakdown Spectroscopy) method as a candidate for diagnostics of tritium measurements in thermonuclear devices.

The research was also focused on investigation of the effects of the laser irradiation on morphology and microstructure of deposits and substrates. The morphology of the sample before and after laser irradiation was characterized by scanning transmission electron microscopy (STEM), high resolution scanning electron microscopy (HRSEM), focus ion beam (FIB) and energy-dispersive X-ray spectroscopy (EDS).

P2-005 **Stability of liquid metal thin film flow under varying heat load**

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Liquid metals form a promising first wall candidate for divertors in future fusion reactors. However, the integrity of the metal film needs to be ensured under all circumstances and therefore the stability of such a flow needs to be addressed. Previous work by Gao and Morley [Magnetohydrodynamics 38, No. 4, 359-375, 2002] has identified the stability limits of a thin film metal flow in a spanwise varying magnetic field, but did not include the effect of temperature dependent material properties like density, viscosity, surface tension and electrical conductivity. These effects need to be taken into account because of the extreme heat loads on the divertor wall. In our analysis, we determine the flow profile and stability limits for a thin film metal flow down an inclined surface under a spatially varying heat load, using temperature dependent material properties.

P2-006 **Liquid Film First Wall Feasibility**

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Stability of the open free surface of liquid metal film in vacuum is of interests for some fusion application. This study reveals inherent stability of the film on the curved wall that can be applied for the ICF chamber, or IFMIF-type liquid metal target from the wave stability of the flowing liquid metal film on a cylindrical surface. A stability of the thin liquid metal film which flowing inside and outside of the cylindrical surface under vacuum or with low pressure ambient gas is considered. For the analysis, a simplified formula to describe the stability condition is deduced from the Navier-Stokes equation and boundary conditions at first order approximation. Gravity and viscosity factors are neglected and only surface tension force is considered by the non-dimensional number comparison. In case of inner cylindrical surface flow, result shows it is almost stable except for the very high density ambient gas condition. Stable condition is described as (n / We) > Q2 ; azimuthal wave number divided by Weber number is greater than the density ratio (density of ambient gas per liquid density). This is the identical result with the stability of the sinuous mode straight free fall liquid film. On the contrary, in case of the external cylindrical surface flow, it is always unstable due to the opposite effect of the Laplace's difference pressure, and causes the azimuthal plane instability waves. It is independent from the density of the ambient gas condition. For the experimental verification, a device to observe the straight free fall thin liquid film is fabricated. As a liquid media, Pb-17Li is used.

P2-008 **Effect of induced damage on hydrogen isotope retention of F82H with impurity layer**

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For the fusion reactor device, various kinds of particles with various energies are incident on the first wall. These incidences result in the formation of surface damage and/or impurity layer on the surface of the first wall material. These must affect on the retention of implanted particles such as hydrogen isotope used as the fuel of fusion reaction. The tritium inventory is an important issue for safety of fusion reactor devices. On the other hands, some implanted particles are desorbed from the surface of the first wall. The desorption behaviors are connected with the development of effective wall conditioning method in the fusion reactor devices. So, it is necessary to elucidate of the relation between characteristics of the surface layer such as damage and impurity distributions and hydrogen isotope retention/desorption. In the present study, the relations between the characteristics of the surface with damage and impurity and the hydrogen isotope retention/desorption behavior for F82H have been investigated with plasma irradiation apparatus. The impurity layer was formed on the surface of F82H by plasma/ion exposure. Then, the samples were irradiated with inert gas ions. In the irradiation experiment, the ion energy and fluence was changed. After the irradiation, deuterium was implanted into the samples. After the deuterium irradiation, the samples were analyzed by using thermal desorption spectroscopy, Auger electron spectroscopy and electron microscopy. The obtained results clearly showed the induced damage significant influence on the deuterium retention/desorption properties. The effect of the impurity layer will be also presented.

P2-009 **Calorimetry and Electron Beam Control in Korea Heat Load Test Facility KoHLT-EB**

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Korea Heat Load Test facility, KoHLT-EB (Electron Beam) has been operating for the plasma facing components to develop the fusion engineering in Korea. The ITER Neutral Beam Duct Liner (NBDL) was fabricated and tested to qualify the thermocouple fixation method for the temperature measurement during direct collision of the high-power neutral beam during ITER operation. The NBDL is CuCrZr panels which are actively water cooled using deep drilled channels. In order to perform the profile test, the assessment for the possibility of electron beam Gaussian power density profile and result of absorbed power for that profile before the test start must be need. To assess the possibility of Gaussian profile, for the qualification test of Gaussian heat load profile, small calorimetry was manufactured to simulate real heat profile in the neutral beam duct liner, and this calorimetry has two cooling channel five thermocouples, same as NBDL. Preliminary analyses with ANSYS-CFX using a 3D model were performed with the calorimetry model. The heating area was modeled to be 60 mm x 250 mm. The simulated heat flux is 0.5 - 2.2 MW/m^2 at with 0.75 kg/sec of water flow rate. A steady heat flux test was performed to measure the surface heat flux, surface temperature profile. With a thermo-hydraulic analysis and heat load test, the Gaussian heat profile will be confirmed for this calorimetry and NBDL mockup. The Korean heat load test facility will be used to qualify the specifications of various plasma facing components in fusion devices.

P2-010 **Underwater explosive welding of tungsten and reduced-activation ferritic steel F82H**

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The blanket first wall and divertor using tungsten-coated ferritic-martensitic(RAFM) steels are key technologies to realize the fusion reactor because these components will be used in severe environments with energetic particle irradiation at high temperatures. Therefore, various coating/welding technologies have been examined: diffusion bonding [1], brazing [1], and vacuum plasma spray [2]. However, it is understood that mechanical properties in the vicinity of the bonding interface are deteriorated due to the thermal stress induced by the difference in the coefficients of thermal expansion of tungsten and RAFM at high temperatures during the processing. The present study reports the underwater explosive welding of commercial pure tungsten on a RAFM steel F82H. We examine the underwater explosive welding which has been developed by one of the authors [3]. Using this technique, the welding of difficult to weld materials such as a thin metal plate on ceramic and the welding of tungsten on a copper base plate have been demonstrated. As a result, Cross-sectional images and chemical compositional maps were obtained using EPMA. The underwater explosive welding was successfully used to clad tungsten on ferritic steels. Cross-sectional image represents a wavy surface, a thin inner layer was confirmed. The compositional analysis on the interface region shows formation of a thin mixed layer of W and Fe. We will discuss the W

[1] N. Oono et al., Journal of Nuclear Materials 417 (2011) 253.
[2] T. Nagasaka et al., Journal of Nuclear Materials 417 (2011) 306.
[3] K. Hokamot et al., Journal of Materials Processing Technology 85 (1999) 175.

P2-011 **Progress in the design of Normal Heat Flux (NHF) First Wall panels**

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The design of ITER blanket is currently progressing towards the final stage of completion and particular attention is devoted to the design of First Wall (FW) panels, which are among the most important components of the blanket system, directly facing the plasma. According to the operation modes of the ITER reactor, two different categories of FW panels are considered: Enhanced Heat Flux (EHF) and Normal Heat Flux (NHF), which are able to withstand heat flux up to 4.7 MW/m2 and 2 MW/m2, respectively.

F4E and IO are working towards the Final Design Review of the 218 NHF FW panels, which represent the EU commitment to the construction of ITER blanket. A typical NHF FW panel consists of a series of fingers, which can be considered as the elementary plasma facing units. A finger is made of three different materials, stainless steel for the supporting structure, copper chromium zirconium for the heat sink and beryllium as armour material.

This paper presents the assessment of the NHF FW modules placed in the blanket rows 2, 6, 12 and 18. Finite Element (FE) models have been developed and numerical simulations have been conducted to predict the behaviour of the whole FW panels. Subsequently, on the basis of the results of the global models, detailed sub-models have been defined for the most critical fingers and corresponding sections of the supporting beam. The global and detailed models were built using ANSYS and ABAQUS codes, including more than 2 million elements. Results of the FEM simulations both on the panel and finger models show that the current design of FW is able to withstand the loading conditions during ITER operation. Local design changes are implemented in order to solve stress concentration issues and to increase design standardization in the FW components, towards the Final Design Review.

P2-012 **Characterization of HIP joints for ITER First Wall involving SS, Cu, CuCrZr and Be**

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The First Wall (FW) Panels of the ITER blanket consist of a complex bimetallic structure of 316L stainless steel (SS) backing plate and a copper alloy (CuCrZr) heat sink, covered with Beryllium armour tiles. Joining of these materials will be done by HIP, following the solid state diffusion bonding technique. This work shows studies on joints involving similar and dissimilar metal bonds of SS, Cu, CuCrZr and Be produced by HIP according to ITER recommendations for FW Panels manufacturing. SS/SS, Cu/Cu, CuCrZr/CuCrZr and SS/Cu joints were obtained after HIP at 1040ºC, whereas a lower temperature HIP treatment at 560-580ºC was performed to join Be to CuCrZr. Since fast cooling after HIP was not feasible, a subsequent solution annealing treatment at ~980ºC followed by gas quenching (cooling rate > 1.8 º/s above 450ºC) was adopted in the systems involving CuCrZr. CuCrZr/CuCrZr joints were subjected to a final ageing heat treatment at ~580ºC to study the precipitation hardening of CuCrZr during the final HIP treatment. Additionally, different coatings of Cu, Cr and Ti were applied on Be by PVD before HIP to act as diffusion barrier between Be and CuCrZr and prevent the presence of brittle beryllides. Best results were obtained with a coating of Ti (5 microns) followed by Cu (10 microns). The influence of the solution annealing treatment and further ageing of CuCrZr during HIP at ~580ºC on its microstructural degradation in terms of grain growth and precipitates coarsening was evaluated. Non-Destructive Examination by Ultrasonic Technique (UT) was performed after HIP to detect defects produced at the interface. After HIP and subsequent thermal treatments the microstructure of the joints was analyzed by microscopy and compared with UT results.

P2-014 **A solution for operation embrittlement of tungsten components: tungsten fibre-reinforced tungsten**

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Tungsten is – due to a unique combination of properties – a candidate for the plasma-facing material in a future fusion reactor. The use of tungsten is not limited by its strength but rather by its lack of damage tolerance. The lower operation limit is the ductile-to-brittle transition temperature which is typically between 520 and 920K. The upper limit is the recrystallization temperature of about 1570K where tungsten becomes very brittle. This problem is exacerbated by embrittlement during operation due to intense neutron irradiation from the D-T fusion reaction. With tungsten fibre-reinforced tungsten composites (Wf/W) we developed a new composite material combining high strength and ductility in the fibres with extrinsic toughening mechanisms. The toughening effect is effective in the as-produced as well as in the embrittled state, since extrinsic toughening also works for fully brittle composite components. A newly developed manufacturing method based on the chemical vapour infiltration of tungsten allows the fabrication of Wf/W composites at temperatures below 970K and without mechanical loads.

The toughening mechanisms were proven to withstand embrittlement by means of mechanical tests on miniaturized samples. To embrittle the samples they were annealed at temperatures of 2000K for 30min. A newly designed method for 4-point bending testing in combination with high energy synchrotron tomography allowed the validation of active toughening mechanisms. In addition, bending tests in combination with in-situ observation in an electron microscope were used to get a first idea about the quantitative contribution. Although different mechanisms of toughening were found to be active compared to the as-produced case, the toughening is still active. The results give reasonable hope that Wf/W can broaden the operation window for the use of tungsten and to solve the problem of neutron embrittlement.

P2-015 **Progress in Development and Application of Lithium Based Components for Tokamak**

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Experiments with lithium plasma facing components (PFC) show promising results in operation of fusion facilities and improvement of it plasma parameters. Our activity on development of new lithium based plasma facing material (PFM), PFC design and technology of lithium application in tokamak is addressed to solution of problems of steady-state tokamak reactor. Progress in development of limiters with different kind of lithium capillary porous system (CPS) for T-11 tokamak is considered in terms of protection of tokamak PFC from damage under normal operation, ELMs and disruptions, prevention of overheating during hot plasma interaction, elimination of plasma pollution and lithium accumulation in tokamak chamber. Overview of design and result of experimental tests of the liquid lithium limiters on the base of CPS from stainless steel, molybdenum and tungsten are presented for the following versions: horizontal rail type limiters with passive and active cooling; vertical limiter for demonstration of possibility to agenize the closed lithium circulation in tokamak chamber. Experiments with presented lithium CPS based limiters in T-11M tokamak conditions have demonstrated improvement in plasma parameters, limiter proper operation, good resistance of lithium CPS to damage, stability of liquid lithium surface and possibility of self-renewal. Structure and result of application of ring CPS based limiter for investigation of lithium behavior in tokamak chamber is presented. The first result of application of technique for lithium removal from the tokamak chamber is considered and structure of applied cryogenic tool is presented. Results of this activity create the base for development of lithium PFC under the Russian project of volumetric neutron source.

P2-016 **High heat flux PLIF facility for the first wall component test**

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In the tokamak device, the first wall component is a typical plasma facing component (PFC), and there is high heat flux from the plasma onto the surface of the PFC, which could rise to several MW per square meters; the high temperature and huge thermal stress due to this environment is a critical danger to the component, and strong cooling method is being developed to protect the first wall; boiling concepts like hypervapotron and swirl tube are two kinds of advanced water cooling techniques. Because there is complicated two-phase flow in these boiling concepts, it is necessary to check the flow status to verify the cooling performance of the design, and high speed camera and planar laser induced fluorescence (PLIF) with proven dye could clearly show both the turbulent streamline and temperature contour in a wide range, which gives a global view and is much more useful than dispersive sensors; meanwhile, in order to reach a high heat flux load for the PFC test, electron beam and neutron beam are the main types and cost much to build a huge facility; for easy and daily experiment requirement, an induction heating module is employed to produce 1MW/m2 heat flux on the first wall mock-up heating surface. These efforts lead to an easy way of high heat flux component testing; this paper takes a facility with a hypervapotron mock-up as an instance.

P2-017 **Manufacture of 14 First Wall Mock-ups with calibrated defects**

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This paper describes the fabrication of reduced scale ITER First Wall (FW) mock-ups of the Normal Heat Flux (NHF) design. The activity was carried out in support to qualification of the European Domestic Agency for the supply of the European share of the ITER first wall. Main purpose was to manufacture mock-ups to be tested in a High heat Flux test facility to define defects acceptance criteria. To this goal, First Wall (FW) Mock-Ups (MUs) with calibrated defects were designed and manufactured. Several kinds of defects have been considered, depending on their position (Be/CuCrZr joint corners, Be/CuCrZr joint sides, CuCrZr/CuCrZr flat joint, CuCrZr/316L SS flat joint, SS pipes/CuCrZr cylindrical joint) and geometry (flat rectangle, flat triangle, cylindrical segment surface), and implemented on a set of fourteen mock-ups. The paper describes, more specifically, the fabrication of the calibrated defects with Alumina and quartz inserts at the different interfaces, the results obtained at the end of the manufacturing process, with the description of the most significant steps and the lessons learned. The results of the ultrasonic testing performed at the end of the manufacturing will be also presented, including a description of the UT equipment used to perform the inspection of the cylindrical defects by the inner side of the Stainless Steel tubes.

P2-018 **High Heat Flux Test of a Korea ITER TBM FW Mock-up**

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Korea has developed two Test Blanket Modules (TBMs) for participating the ITER TBM program; a Helium Cooled Solid Breeder (HCSB) TBM and a Helium Cooled Molten Lithium (HCML) TBM, respectively. Recently, solid type TBM, HCSB was decided to be tested in the ITER and the name was changed to a Helium Cooled Ceramic Reflector (HCCR) considering the unique concept of using the graphite reflector. The newly designed HCCR TBM has four sub-modules and a sub-module is a box with a rectangular structure with a faceted first wall (FW). The front surface of the sub-module is 231 mm in width and 835 mm in height. In the FW of the sub-module, there are rectangular shape of 9 cooling channels with 15 mm in width and 11 mm in height. Before the HCCR TBM was chosen as a leading concept in Korea for the ITER TBM program, a 1/6-scale mock-up of the TBM FW has been fabricated under the HCML TBM project. The size of the mock-up is 260 mm height and 444 mm width and there are rectangular shape of 10 channels with 20 mm width and 10 mm height. It is similar to a half size of the sub-module of the HCCR TBM and the fabrication methods development has been continued. To develop the fabrication method for the TBM FW and verify its integrity, a half-scale sub-module mock-up was fabricated and integrity test has been prepared. A mock-up was assembled by HIPping of the previous fabricated components between the welded front and back plates, and then machining these plates to form the completed sub-module FW. For confirming the joint integrity of the fabricated mock-up, it has been tested at the E-beam high heat flux (HHF) test facility.

P2-019 **Current Status of Structural Components Design of a Korean HCCR TBM in ITER**

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Korean has designed a Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM) to be tested in ITER. The design of the main structural components susch as First Wall (FW) ans Side Wall (SW) was obtained through the optimization process to meet the design requirement. For FW, structural integrity was evaluated for internal pressure of 10 MPa He coolent and thermal expansion effect among four sub-modules. For SW, structural integrity by internal pressure in the same conditions to the FW and flow distribution function were evaluated. Through the design and performance analysis, the current design of the FW and SW in KO HCCR TBM shows that they meet the design reirement and keep their function of flow distribution and cooling.

P2-020 **Comparison of hydrogen isotope retention for tungsten probes in LHD vacuum vessel during the experimental campaigns in 2011 and 2012**

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In the future fusion reactor, tungsten will be used as a plasma facing material. One of the key issues for the usage of tungsten as a plasma facing material is the tritium retention evaluation under actual fusion conditions. In ITER, the tritium retention may be underestimated if the tritium retention in the impurity deposition layer is not considered. It is important to predict actual tritium retention in plasma facing materials. This paper presents the comparison of hydrogen isotope retention enhancement for tungsten exposed to long-term plasma campaigns in 2011 and 2012 at LHD. In 2012 campaign, the closed divertor structure, which consists of graphite parts, was installed in 8 sections to enhance the plasma performance, although that in 2011 campaign was 2 sections. The mirror finished disk-type tungsten samples were placed into in three or four typical positions, namely the higher plasma wall interaction area (PI), deposition area (DP), higher heat load area (HL) and erosion dominated area (ER). These samples were exposed to ~ 6700 shots of hydrogen plasma for 2011 campaign and ~ 5000�@shots for 2012 campaign. The sample temperature was less than 373 K during the exposure. Thereafter, the 1.0 keV deuterium ions were additionally implanted into these samples with the flux of 1.0�~1018 D+ m-2 s-1 up to the fluence of 5.0�~1021 D+ m-2 to evaluate the enhancement of hydrogen isotope retention capacity. The hydrogen isotope retention was estimated by Thermal Desorption Spectroscopy (TDS). The Glow Discharge-Optical Emission Spectroscopy was also performed to study the depth profiles of constituent atoms.

The D2 TDS spectrum for PI in 2011 campaign consisted of three desorption stages at ~ 600, 800 and 1000 K, and large D desorption was observed at 800 K. This behavior was quite different from that for 1.0 keV D2+ implanted pure tungsten, where most of deuterium was desorbed less than 600 K. In the case of 16th campaign sample, large D2 desorption stage was shifted to 900 K, indicating that the chemical state of carbon impurity layer was changed and crystallization like graphite structure would be proceeded. The thickness of impurity layer for 2012 campaign sample was thickened and large H and D were observed. These results indicate that the chemical structure for carbon impurity layer may control H and D retention enhancement.

P2-021 **Current Status of Plasma Facing Components for the WEST Project**

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The WEST (W -for tungsten- Environment in Steady-state Tokamak) project is based on an upgrade of Tore Supra, moving from a carbon limiter to an X-point tungsten divertor device, while keeping advantage of its long discharge capability. This is obtained by inserting in vessel coils to create the X-point while adapting the in-vessel components to this new (plasma) configuration. As in any tokamak, plasma-facing components (PFCs) must provide adequate protection of in-vessel structures, sufficient heat exhaust capability and be compatible with the requirement of plasma purity. These functions take on particular significance for the WEST project since as in ITER, it will combine long pulse (>1000s), high power operation (up to 15 MW for WEST) with severe restrictions on permitted core impurity concentrations and which, in addition, will produce transient energy loads. In this context, WEST will offer the key capability of testing relevant ITER technologies of high heat flux actively cooled components in real plasma conditions, in particular the challenging tungsten divertor. The current WEST PFCs design has now reached a mature stage following the 2012 WEST Conceptual Design Review. The PFCs will be fully metallic and actively cooled, with ITER tungsten (W) monoblocks technology for the lower divertor, and metallic cover W or B (by deposited coating) for the others in vessel components.

In this paper the main features of WEST PFCs are presented including the key elements of the design according to the physics specifications (essentially thermal load specifications) used to define the different concepts, the material selection, technologies and design issues. Additionally, the current status concerning the feasibility studies, started with industrial companies about the manufacturing (assembling technology, fabrication processes, etc…) of the main actively cooled components, will be also reported.

P2-022 **Thermo-mechanical analysis of RMP coil system for EAST tokamak**

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Resonant magnetic perturbation (RMP) has been proved to be efficient approach on edge localized mode control, resistive wall mode control and error field correction, RMP coil system under design in EAST tokamak will realize above-mentioned multi-functions. This paper focuses on the thermo-hydraulic and thermal-structural analysis of EAST RMP coil system on basis of sensitivity analysis, both normal and off-normal (baking out of passive plate is foreseen) working conditions are considered. The most characteristic set of coil system is chosen with a complete modeling by means of 3D FE method, thermal-structural performance is investigated adequately, both locally and globally. The compromise is made between thermal performance and structural design requirements, and the results indicate the optimized design is feasible and reasonable.

P2-023 **Design of a monoblock type water cooled DEMO divertor using Eurofer as structural material.**

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This paper presents performed investigation aimed to optimise the design of a monoblock type water cooled DEMO divertor using Eurofer as structural material and tungsten as armour material, in order to withstand heat flux up to 10 MWm-2. A parametric analysis has been carried out at first in order to assess the influence of geometrical and thermo-hydraulic parameters on the monoblock thermomechanical behaviour. An "optimised" geometry has been identified featuring a Eurofer pipe with 6 mm inner diameter. A second design has also been assessed featuring the same geometrical inner diameter as the ITER reference monoblock, i.e. 12 mm inner diameter. Thermal hydraulic conditions have been chosen in order to meet the Eurofer temperature operating range (325°C – 550°C). Water pressure conditions of pressurized water reactor has been assumed (pin = 15.5 MPa) for the coolant. The water velocity has been chosen in order to guarantee a margin of 1.2 to the critical heat flux without excessively pressure drops (V=20ms-1). The possibility of castellating the first mm of the W exposed plasma surface (2.5 over 3.5 mm) has also been assessed as possible solution to reduce stresses in Eurofer. 3D thermomechanical analyses assuming "degraded" (under irradiation) and "non-degraded" material properties have been performed and RCCMRx design criteria have been checked.

Performed analyses showed that a design concept of a monoblock type water cooled divertor using Eurofer as structural material could be suitable for an incident heat flux up to 10 MWm-2. The progressive deformation and fatigue are the limiting damages, as expected, and relevant criteria for 5200 cycles of ~2hours are not met. Monotonic type damages do not seem to be an issue, even when embrittlement under irradiation is considered. Finally, castellations improve thermomechanical behaviour; nonetheless their technological feasibility and their possible role of crack initiator should be furthermore investigated.

P2-024 **Mechanical properties of the advanced tungsten alloys**

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Tungsten (W) is a candidate for armor material in divertors and first walls in fusion reactors because of its high melting point, thermal conductivity, and sputtering resistance. During the operation of a fusion reactor, armor materials are exposed to high-energy neutron irradiation and high heat flux, and irradiation embrittlement and/or recrystallization embrittlement will be occur. The irradiation embrittlement and recrystallization embrittlement causes the degradation of ductility and strength in tungsten. Alloying, refining grain structures and dispersing second phase are the typical ways to suppress the irradiation embrittlement and recrystallization embrittlement. Microstructures and mechanical properties of pure W, W–Re alloys, K–doped W, and La–doped W were investigated in this work. W–Re alloys and K or La–doped W shows the high recrystallization temperature and bending strength compared with pure W. Furthermore, we fabricated the new tungsten alloys such as K–doped W–3%Re and La–doped W–3%Re. This paper presents the effect of multiple additions on microstructure and mechanical property changes in pure W.

P2-025 **Experimental Try-out of IR Thermography Method for Final Acceptance Tests of the ITER Divertor Dome**

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The divertor dome (DO), being part of the ITER divertor, is designed to evacuate the major part of the plasma thermal energy. As a plasma-facing component, the DO experiences high heat fluxes (up to 5.0 MW/m2). Such severe operation conditions of the DO imply stringent requirements for the DO design and its cooling system to ensure the required temperature operation regime of the dome. Hence, Final Acceptance Tests (FAT) shall be performed on each DO final assembled component with the aim to demonstrate that none of parallel coolant channels are blocked or partially blocked. The paper presents the results of the analytical and experimental validation of the thermography method to be applied for the FAT to determine defective hypervapotrons of the divertor dome based on contactless measurement of the dynamic temperature field of the PFC surface at a steplike increase (from zero to constant value) in the coolant flow rate with a temperature higher than that of the hypervapotron.

The accuracy of the thermography method is the higher the steeper is the temperature front of water delivered to the inlet of cooling channels. As DOs consist of three cooled-in-series PFCs, more diffused temperature front arrives at each successive PFC, thereby the applicability of the analyzed method for the DO FAT is called into question. The tests have been carried on the mock-ups representing all three components of the DO and in a wide range of coolant flow rates and initial temperatures. The performed study reveals the fair dependence of the dynamic characteristics of the PFC surface temperature on the coolant flow rate even for the third consecutive PFC. Thus, the workability of the proposed method for detection of the partially blocked channel has been verified.

P2-026 **Interaction of tritium and helium with lithium -lead eutectic under reactor irradiation**

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It is well known that plumbum-lithium eutectic is a prospective material for liquid blankets of future fusion reactors. One feature of these blankets is tritium reproduction for its future use as a fuel of fusion reactor. For correct estimation of tritium quantity in a blanket it is necessary to know about tritium diffusion in the plumbum-lithium eutectic under the conditions similar to real operation that is under neutron irradiation. At present, it is difficult to create such the conditions of thermo-nuclear neutron irradiation (energy of 14 MeV): the existing facilities (neutron generators and tokamaks) cannot provide for sufficient fluxes and fluences of thermonuclear neutrons, and in-situ experiments require the development of new techniques and facilities. Thus, currently the most rational solution is reactor studies for assessment influence of neutron irradiation on the parameters of tritium interaction with plumbum-lithium eutectic. Our studies were carried out at the reactor IVG1.M, Institute of Atomic Energy, Kurchatov, Kazakhstan. The first stage of the experiments included absorption studies; there were obtained temperature dependencies of rate constant of hydrogen isotopes interaction with plumbum-lithium eutectic under various reactor powers. The obtained dependencies were used for determination of main interaction parameters such as the activation energies and Arrenius pre-exponents. Second stage of the experiments included mass-spectrometer studies to measure tritium and helium release from plumbum-lithium eutectic under various temperatures and reactor powers. These dependencies were used to determine the efficiency of tritium and helium release from plumbum-lithium eutectic under various experimental conditions. The study results indicated a significant increase of diffusion and mass-transfer in plumbum-lithium eutectic under reactor irradiation.

P2-027 **Development of Lithium meta-titanate Ceramics pebbles for Indian LLCB TBM**

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The Indian Lead-Lithium Ceramic Breeder (LLCB) Test Blanket Module (TBM) is the Indian DEMO relevant blanket module which will be tested in ITER machine through the TBM program. Lithium meta titanate (Li_2TiO_3) will be used in Indian Lead Lithium Ceramic Breeder (LLCB) concept to be tested in ITER. Li_2TiO_3 is being considered as one of the promising tritium breeding materials for future DEMO reactors because of its reasonable lithium atom density, prominent tritium release rate at low temperatures, its low activation characteristics, low thermal expansion coefficient & high thermal conductivity etc. Among the various methods of preparation of Li_2TiO_3 powders, Indian TBM team is involved in developing this tritium breeder material by solid state reaction and solution combustion method. Lithium carbonate (Li_2CO_3) and titanium di-oxide (TiO_2) are used as a raw material for solid state methods whereas TiO_2 was dissolved to make $TiO(NO_3)_2$ for solution combustion method. The final powder has been characterized for its phase purity, grain size and its true density with XRD, SEM & with Helium-Pycnometer. Finally Li_2TiO_3 pebbles were prepared by the extrusion followed by the spheronization with diameter range from 0.8 mm to 1.5 mm. Prepared pebbles were tested for mechanical strength using crush strength measurement, pebble size distribution and pore size distribution and their analysis by Mercury porosimeter and Archimedes principle. The details of the powder synthesization, pebble formation and their various characterizations will be discussed in this paper.

P2-028 **Measurements of the purge helium pressure drop across pebble beds packed with lithium orthosilicate and glass pebbles**

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The Helium Cooled Pebble Bed (HCPB) blanket contains pebble beds of lithium orthosilicate for tritium breeding and beryllium for neutron multiplying. These pebble beds are purged by helium to transport the bred tritium to the tritium extraction system. The pressure drop of the purge helium has a direct impact on the relevant pumping power. Therefore the objective of this study is to present reliable measurements of the purge helium pressure drop across different pebble beds packed with: (i) lithium orthosilicate pebbles with a diameter range of 0.25–0.49 mm and (ii) glass pebbles with four diameter ranges of 0.25–0.5, 0.5–0.75, 0.75–1.0, 0.9–1.2 mm. The pebble bed was formed by packing the pebbles into a cylindrical stainless steel container and it had a diameter of 30 mm and a length of 120 mm. The pebble bed was integrated into a gas loop that has four serial variable-speed side-channel compressors to regulate the helium mass flow. To determine the pressure drop across the pebble bed, the static pressure was measured at two locations (100 mm apart) along the pebble bed and at the inlet and outlet locations. The obtained results demonstrated that: (i) the pressure drop significantly increases with decreasing the pebbles' diameter, (ii) for the same superficial velocity, the pressure drop is directly proportional to the inlet pressure, and (iii) theoretical predictions of Ergun's equation agree well with the experimental values of the pressure drop. In addition, the measured values of the pressure drop for the lithium orthosilicate pebble bed will support the design of the purge gas system for the HCPB breeder units.

P2-029 **A Neutron Poison Tritium Breeding Controller Applied to a HCPB Fusion Reactor Model**

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The generation of tritium in sufficient quantities is an absolute requirement for a next step fusion device such as DEMO due to the scarcity of tritium supplies. A number of methods have been proposed in order to meet this requirement, however a lithium-based tritium breeding blanket that surrounds the fusion plasma is widely considered to be the most suitable solution. Although the production of sufficient quantities of tritium will be one of the main challenges for DEMO, within an energy economy featuring several fusion power plants the active control of tritium production may be required in order to manage surplus tritium inventories on power plant sites. The primary reason for controlling the tritium inventory in such an economy would therefore be to minimise the risk and storage costs associated with large quantities of surplus tritium. In order to ensure enough tritium will be produced in a reactor which contains a solid tritium breeder, over the reactor's lifetime, the tritium breeding rate at the beginning of its lifetime is relatively high and reduces over time. This causes a large surplus tritium inventory to build up until approximately halfway through the lifetime of the blanket, when the inventory begins to decrease. This tritium inventory could exceed several tens of kilograms of tritium, impacting on possible safety and licensing conditions that may exist. This paper describes a possible solution to the tritium inventory problem that includes neutron poison injection, which is managed with a tritium breeding controller. A PID controller is used to inject neutron absorbing compounds into the helium coolant, depending on the difference between the required tritium excess inventory and the measured tritium excess inventory. The compounds effectively reduce the amount of neutrons available to react with lithium compounds, thus reducing the tritium breeding ratio.

Previous tritium breeding studies have shown that coupled neutron-transport and burn-up is required in order to accurately predict the tritium self-sufficiency. The FATI code (Fusion Activation and Transport Interface) will be used to perform the tritium breeding and controller calculations. FATI couples MCNP and FISPACT-II in a cyclic manner and implements controller mechanisms between transport and burn-up steps. Results are presented from the FATI code applied to a CCFE HCPB DEMO conceptual model.

P2-030 **Optimised mass flow rate distribution analysis for cooling the ITER Blanket System**

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A number of design evolutions since the 2007 ITER Design Review have justified a revisit of the water mass flow rate distribution to the Blanket System. This paper presents the rationale to the optimisation of water distribution in ITER blanket modules (BM), meeting both blanket system requirement and compliant with interfaces. The water mass flow rate distribution needs to be compatible with several constrains. The key ones are the following: be compatible with the overall water allocation (3140 kg/s for 440 wall mounted BMs), meet the critical heat flux margin of 1.4 in the plasma facing units, meet a maximum temperature increase of 70°C at the outlet of each single BM, ensure that water velocity is less than 7 m/s in all manifolds, and that the pressure drops of all BMs can be equilibrated. The methodology is presented. Several calculation steps are included like:

•Critical heat flux margin is checked individually for each individual first wall panel to ensure the required margin. It is checked with experimental results obtained with mock-ups working at ITER conditions.

•Nuclear and surface heating loads have been reviewed and applied in each module following self-consistent accounting of the global power balance.

•Non-regular NBI sectors are included on the study, considering their individual particularities (module size, re-ionisation heat load).

In spite of the challenging constraints due to the large variety of modules to be cooled and also due to the number of requirements and interfaces of the Blanket System, the proposed methodology allows finding an optimised water distribution meeting all requirements. The obtained mass flow rate leaves no un-used margin for normal operating conditions, indicating that the optimization is complete, and showing that global water allocation is at the minimum possible. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

P2-031 **Simple Thought Experiment to Examine Benchmark Performance for Fusion Nuclear Data**

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Many nuclear-data measurements (differential experiments) were carried out so far especially fort fusion reactor. There are similarly many integral benchmark experiments to check the nuclear data. The differential experiments can be used for a direct test of the nuclear data at 14 MeV, because they were mostly conducted with a 14 MeV neutron source. Integral experiments carried out with the 14 MeV neutron source were commonly but vaguely expected to become a test also for energies below 14 MeV. The author�fs group thus began to investigate how efficiently the integral benchmark experiments can work also as a test of nuclear data for MeV energy region. In this study, we focus on leakage-neutron spectrum measurement. In the measurement, neutrons transporting in a rectangular sample are finally measured by a detector. A thought experiment was performed to examine for what energy of neutron the detected neutron contribution could benchmark especially in case of Monte Carlo transport calculations. Effects appearing in the measured spectrum were examined when the cross sections were artificially changed to be a little smaller or larger. The results can estimate the reason of the discrepancy between the measured and calculated spectra. From the result, for the F5(point detector) tally, contribution measured by the detector originates not only from a neutron making the contribution but also a previous neutron making the neutron, which made the contribution. In conclusion, detector contribution naturally cannot be used for benchmarking of a neutron(A) �gconveying�h the contribution to the detector, but of course can be used for benchmarking of a neutron(B) �gmaking�h the contribution. However, if a neutron(C) producing the neutron(B) exists, the detector contribution will also be available for benchmarking of the neutron(C). Moreover, it is surprisingly the same how efficiently neutrons(B) and (C) contribute to benchmarking. This might suggest benchmarking possibility without adjoint calculations.

P2-032 **Mechanical Behaviors of FCI in Thermo-Magneto-Structure Coupling Field**

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Fusion blanket module plays an important role in ITER, in which structures (flow channel insert, first wall, etc.) are subjected to strong magnetic field and high temperature field. In this study, the dynamics behaviors of the segment are investigated. Moving structure subjected to external load and magnetic field will result in circular electric current and Lorentz force, which will affect dynamic responses of the structure. Lorentz force produced by both external and induced magnetic field was taken into consideration in this work. Finite element method on basis of both solid mechanics and electromagnetics is applied to analyze the temperature field and the dynamics behavior of structure. APDL and macro code was used to re-develop magnetic-structure coupling solving capacity of ANSYS. Finally, the effects of temperature gradient and induced magnetic field and influence of thermal conductivity and electric conductivity of structure material are presented.

P2-033 **Electrical Connectors for Blanket Modules in ITER**

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Blanket electrical connectors ("e-straps") are low-impedance electrical bridges crossing a gap between blanket modules ("BMs") and vacuum vessel (VV). Similar e-straps are used between two components of each BM: the first wall panel and the shield block. The main functions of E-straps are to: (a) conduct halo currents intercepting some rows of BM, (b) provide grounding paths for all BMs, and (c) operate as electrical shunts which protect water cooling branch pipes from excessive halo and eddy currents. E-straps should be elastic enough to absorb 3-D imposed displacements of BM relative VV in a scale of +/-2 mm and at the same time strong enough to not be damaged by induced EM loads. Each electrical strap is a package of flexible conductive plates made of CuCrZr bronze. Halo current up to 130 kA and some components of eddy currents do pass through one E-strap for a few tens or hundreds milliseconds during the plasma vertical displacement events and disruptions. These currents deposit Joule heat and cause rather high electromagnetic loads in a strong external magnetic field, reaching 9 Tesla. A failure of one E-strap gradually redistributes halo and eddy currents into branch pipes and cause their bending by excessive EM loads and Joule heating in accidental electrical contact spots to the surrounding structures, what may lead finally to a water leak.

The paper presents and compares two design options of e-straps: one resembling baseline with several modifications, and another less similar to the baseline. The latter option was developed in 2012 on the basis of more thoughtful analysis of cyclic loading conditions influencing a fatigue lifetime. Detail comparative simulations of current sand fields patterns and subsequent analysis of the fatigue strength and technological assessment allows to make a final recommendation for the e-strap design.

P2-034 **Dynamic Analysis of the ITER Blanket Attachment System**

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ITER blanket attachment system reacts electromagnetic (EM) loads generated in the blanket modules (BM) during the plasma disruptions. EM loads at BM attachments exceed gravity loads by approximately two orders of magnitude. A nearly four-ton BM is attached to the vacuum vessel (VV) by four flexible supports and interfaces with VV shear keys through key pads with assembly gaps of approximately 0.25-0.5 mm at each side. Key pads are enclosed in the blanket modules and serve to protect the ceramic electrical insulation, which is applied at the hidden surfaces of key pads. Module movement relative to the VV results in forced lateral deformations of flexible supports, electrical and hydraulic connectors. These deformations do depend on values of gaps between keys and pads and exceed +/- 2 mm. Dynamic response on EM loading at the keys and flexible supports is rather stochastic. It depends on actual values of the six-eight initial gaps, friction coefficients, initial conditions, and may be very different from quasi-static responses. Dynamic analysis has shown rather large normal and tangential friction forces in open interfaces between VV keys and key pads. The design of some keys has been updated to approximately doubled the load ability based on results of dynamic analysis. The maximum normal and tangent friction forces at key pads, imposed displacements and kinetic energy, obtained with parametric variation of all gaps and for different plasma events, were used as inputs in the relevant experimental program, which is realized in NIKIET and ITER Organization at the stage of a finalization of the blanket design.

P2-035 **Ultrasonic Doppler Experimental Research of Gas Bubble Rising in Liquid Metal under a Strong Magnetic Field**

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Liquid metal is an important energy transport carrier in nuclear fusion reactor. It is important to measure the velocity profile. The method of ultrasonic Doppler velocimetry (UDV) is gradually developed in recent years because of its application in opaque liquid. We experimentally studies the bubble rising in a non transparent liquid metal GaInSn with and without a strong magnetic field in this paper and compared with experiment results of Dresden University Technology.

Firstly, through comparison of the bubble velocities in transparent medium water obtained by high speed video camera and UDV, the accuracy of the UDV is calibrated. We then conduct researches of the bubble rising in GaInSn without magnetic field. The comparison between experimental results with Mendelson equation is illustrated in Fig. 1. Present experimental results in water and in GaInSn, Mori's experimental results in mercury are all close to the Mendelson equation. The forces acting on the bubbles are complicated, such as gravity, buoyancy, surface tension, viscous force and so on which influence the bubble rising trajectory. Akio's drag coefficient expression fits well with the experimental results in water, GaInSn and mercury (1977) in Fig. 2.

Bubble rising in GaInSn under a strong magnetic field is studied. In the experiment, the magnetic induction varies from 0 to 1.97T perpendicular to bubble rising direction. The comparison of the bubble velocities under different magnetic field is shown in Fig. 3. The drag coefficient ratios from present, Mori's and Zhang's experiment results vary with N (interaction parameter), which shows that the greater the interaction parameter is and the smaller the bubble velocity is obtained. The drag coefficient ratios are increased as the interaction parameter is increasing as shown in Fig. 4.

P2-036 **MHD analysis of Lead lithium flow in a duct consisting of circular and square cross-sections under high magnetic field**

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In the Indian Lead Lithium cooled Ceramic Breeder (LLCB) blanket concept, Lead-Lithium (Pb-Li) liquid metal is used to extract heat from its own bulk volume and also from the neighbouring solid breeder zones. The complex flow path of Liquid metal Pb-Li in TBM primarily consist of parallel and anti parallel flow channels, rectangular L/U bends, transition from circular to rectangular regions with different orientation of the applied magnetic field etc. The flow of Pb-Li inside the module will be significantly modified by intense MHD effects due to the presence of high magnetic field. A number of laboratory scale liquid metal MHD experiments are being proposed to understand MHD phenomena in LLCB TBM relevant complex flow geometries and to generate relevant MHD database for validation of numerical codes. In the present work, 3D MHD Numerical flow analysis has been carried out using commercial CFD codes for an experimental test section consisting of circular and square cross-sections with a transition region under applied magnetic field (up to 4T) in a direction perpendicular to flow path. Lead-Lithium at 350 deg C has been used as the fluid and the flow is assumed incompressible and laminar. Analysis of the flow has been carried out for different applied magnetic fields and flow rates. Numerically obtained wall electric potential for both side wall as well as Hartmann wall and pressure drop across the test section have been compared with the experimental data. The liquid metal flow behaviour at various flow regions in the duct has also been studied. The details of the numerical model, analysis procedure and code benchmarking with experimental data will be presented in this paper.

P2-037 **Trapping of deuterium dissolved in fluidized Li by Y**

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Li is one of the attractive materials for liquid blanket, IFMIF liquid target and liquid diverter in fusion reactors. When Li contacts with hydrogen isotopes or is irradiated by neutron, hydrogen isotopes including tritium are dissolved in liquid Li. Since Li has an extremely high affinity for hydrogen isotopes, they are absorbed in Li and are hardly desorbed from Li. When 1 ppm hydrogen is absorbed in Li, its equilibrium pressure is estimated lower than 10^{-10} Pa. Therefore, simple desorption processes such as vacuum evacuation or He purge are not applicable to recover hydrogen isotopes dissolved in Li. Yttrium is a unique metal getter, which has an ability to recover 1 ppm hydrogen isotopes dissolved in Li judging from the solubility data of the Li-H and Y-H systems. However, any experimental proof has never been given. This is because surface oxides interfere hydrogen absorption in Y. In the present study, HF treatment technique is developed to remove surface oxide formed on Y and experiment of deuterium absorption by Y is experimentally investigated using fluidized Li. The HNO_3 and H_2O dissolution techniques are developed to detect hydrogen isotope dissolved in Y and Li, respectively. The reaction is expressed by $YDX+3HNO_3=Y(NO_3)3+HDX$ or $LiDX+H_2O=LiOH+HDX$. The recovery rate of hydrogen isotopes dissolved in fluidized Li by Y is experimentally determined as a function of contact time, absorption temperature and velocity of fluidized Li. The absorption rate is quantitatively compared under various conditions. The experimental results are applied to design an Y trap for recovery of D or T in IFMIF Li flow.

P2-038 **Liquid Metal MHD studies with non-magnetic and ferro-magnetic structural material**

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The structural material for blanket has been proposed to be FMS (Ferritic Martensitic Steel) grade steel material, which is ferromagnetic in nature. In most of the liquid metal MHD experiments, reported so far, SS316/SS304 grade steel has been used as structural material, which is non-magnetic in nature. In a recent experimental campaign, liquid metal MHD experiments have been performed with two identical test sections: one made of SS316L (non-magnetic) and the other one with SS430 (ferro-magnetic).

Hot Pb-Li is used as the process fluid in this experiment. The test section is chosen as a closed square with inlet and outlet at the middle portion of the two horizontal legs, respectively. Pb-Li enters into the test section through a square duct and distributed into two parallel paths through a partition plate. In each parallel path, it travels ~ 0.33 meter length perpendicular to the magnetic field and faces two 90O bends before coming out of the test section through a similar square duct. Experiment has been performed under uniform transverse magnetic field in the range from 0.5 T to 4.0 T. The diagnostics consists of thermocouples for liquid metal temperature measurement, potential pins for wall electric potential measurement and gas based pressure sensors for pressure measurement. The wall electric potential and MHD pressure drop have been compared for the two structural materials under nearly identical experimental conditions. The details of the experiments and experimental results will be presented along with the comparison with analytical results in each case.

P2-039 **Progress in Engineering Design of Indian LLCB TBM Set for testing in ITER**

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The Indian Lead-Lithium Ceramic Breeder (LLCB) Test Blanket Module (TBM) is the Indian DEMO relevant blanket module, as a part of the TBM program in ITER. The LLCB TBM will be tested from the first phase of ITER operation one-half of an ITER port no. 2. . LLCB TBM-Set consists of LLCB TBM module and shield block, which are attached, with the help of attachment system. This LLCB TBM set is inserted in a water-cooled stainless steel frames called ‘TBM Frame’, which also provide the separation between the neighboring TBM-Sets (Chinese TBM set) in port no. 2. In LLCB TBM, the high-pressure helium gas is used to cool the First Wall (FW) structural and lead–lithium eutectic (Pb–Li) flowing separately around the ceramic breeder pebble bed to cool the TBM internals which are heated due to the volumetric heating during plasma operation. Low-pressure helium is purged inside the CB zone to extract the bred tritium. Thermal-structural analyses have been performed independently on LLCB TBM, shield block and also on TBM set as a whole using ANSYS. This paper will also describe the performance analysis of individual components of LLCB TBM set and their different configurations to optimize their performances.

P2-040 **In-situ impedance measurement of corrosion interface in liquid metals**

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Experimental study on in-situ impedance measurement of corrosion interface in liquid metals Pb and Pb-17Li was performed. The purpose of the present study is to investigate the basic corrosion kinetics in the liquid metals and to develop the corrosion monitoring system. Rectangular specimens (10mm x 15mm x 1mm) of an unalloyed iron (Fe) was immersed in the liquid metals Pb and Pb-17Li (30cc) at 773K, and the electro impedance of the wetted surface of the specimen was measured by the impedance measurement system (Princeton applied research, VERSASTAT 3). The surface cross section of the specimen tested was analyzed by EPMA. The oxygen concentration in the liquid Pb and Pb-17Li was controlled at the oxygen saturated condition. The Nyquist plot for the interface between the liquid Pb and the specimen surface was obtained as a single semicircle, and the plot indicated that the interface has a certain electro resistance and a capacitance. However, the shape of the Nyquist plot changed over the immersion time. The electro resistance increased in the initial 20 hours. Then, the electro resistance gradually decreased. These changes of the electro resistance on the specimen surface might be caused by the growth and the breakdown of the oxide layer in the liquid Pb. The results indicated that the surface of the Fe specimen was oxidized by the immersion in the Pb for 1000 hours. The formation of the oxide layer, which had the electrical impedance, was not detected in the test for the Pb-17Li. No formation of the Fe specimen in the Pb-17Li was due to the low oxygen potential of the liquid Pb-17Li. These results indicated that the in-situ impedance measurement method can be applied as the corrosion sensor in the liquid breeders when the corrosion was controlled by the oxidation.

P2-041 **Structural Design and Thermal-Hydraulic Analysis of Liquid Lead-Lithium Tritium Breeder Blanket for China Fusion Engineering Testing Reactor**

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China fusion engineering testing reactor (CFETR) was defined as an ITER-like machine and in good complement with ITER to bridge the gap between ITER and DEMO or fusion power plant in China. Due to the compulsory requirement of tritium self-sufficiency and space limitation in the reactor, tritium breeder blanket design becomes a vitally important technology. In this work, a liquid lead-lithium tritium breeder blanket concept focus on the maintainability, simplicity and feasibility was proposed for CFETR. The modular blanket was expected to be installed and dismantled in the vacuum vessel for reducing the maintenance cost. Two kind of single layer guide tubes were adopted to provide passageway for cutting and welding the pipes and connectors in the blanket. To simplify the blanket structure, the guide tubes were communicated with the first wall flow channel for structure cooling, besides, the tritium breeding zone was divided into several parts for preventing the guide tubes penetrate any cooling plates. Thermal-hydraulic analysis of the blanket was carried out based on the heat load obtained from neutronics calculations and the blanket structural integrity was verified by thermal stress analysis. The preliminary analysis results did not present major feasibility problems.

P2-042 **Influence of surface condition on deuterium release from Li2TiO3 pebble**

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In the blanket of DT fusion reactors, the tritium produced in the tritium breeding material has to be quickly removed for the high tritium recovery efficiency and the minimum tritium inventory. Li2TiO3 is one of the most promising tritium breeding materials, and Li2TiO3 pebbles are employed in a water-cooled solid breeder blanket developed as Test Blanket Module (TBM) in Japan. The tritium is removed by using sweep gas. In the blanket, it is expected that especially the composition of Li2TiO3 pebbles on the surface will be changed by radiation damage, reduction reaction with hydrogen in the sweep gas and so on which might lead to the change in tritium release behavior from Li2TiO3 pebbles. In the present study, the simulation experiments for the tritium release by using the deuterium ion source were conducted. We used the degassed Li2TiO3 pebbles or the pebbles that have radiation damage. These pebbles were irradiated with deuterium ions instead of the tritium generated in the pebbles of the fusion blanket. After the deuterium ion irradiation, the amount of deuterium in the pebbles was measured by using a thermal desorption spectroscopy (TDS). The surface conditions of the samples were also observed by Auger electron spectroscopy. The deuterium irradiated in the pebbles desorbed in forms of HD, D2, HDO and D2O, and the amount of HDO is largest among them during TDS. The shapes of desorption peak of HDO and the amount of retained deuterium were changed for the sample with the radiation damage. The analysis of the surface composition for the damaged sample suggests that the amount of Li near the surface influences the deuterium release behavior. The results for the Li2TiO3 pebbles heated in the hydrogen atmosphere will be also presented.

P2-043 **MHD LiPb Flow and analysis of thermal stress of structure and First Wall in DLL Blanket**

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FDS-II is designated to exploit and evaluate potential attractiveness of pure fusion energy application. The Reduced Activation Ferritic / Martensitic (RAFM) steelstructured He-gas / liquid LiPb dual-cooled (DLL) blanket is one of options of FDS-II. He gas is used to cool the first wall and blanket structure and liquid LiPb is to be the self-cooled tritium breeder with a high outlet temperature up to 700¡æ in order to achieve high thermal efficiency. The Flow Channel Inserts (FCI) are designed and used inside the LiPb coolant channel, which act both as thermal and electrical insulators to keep the temperature of RAFM structure below the maximum allowable temperature.The research is concentrated on the flow and temperature distribution of the both coolants, i.e., LiPb flow and he-gas. The LiPb flow with a transverse magnetic field in vertical channels in the DLL blanket is investigated. It is important for the thermal stress of the structure and first wall to be analysed because of the fluid solid coupling. The simulations of magnetohydrodynamic (MHD) LiPb flow characteristics and of resultant temperature distributions have been performed. The three-dimensional temperature distributions of the LiPb flow in heating duct have been given. The he-gas flow and heat transfer behave are analysed for various velocities and inlet temperatures. The thermal stress of the blanket structure and first wall are studied, which is influenced on the coolant LiPb flow and he-gas flow. The analysis of FCIs stress states based on these thermal loads have been given.

P2-045 **Measurement of Hydrogen Isotope Concentration in Erbium Oxide Coatings**

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Tritium permeation through structural materials in fusion reactors is a serious problem for the establishment of radiological safety and fuel cycle. Tritium permeation barrier (TPB) has been investigated as a promising solution to suppress the tritium permeation. Ceramic coatings have been fabricated on structural materials as TPBs, and their hydrogen isotope permeabilities have been evaluated for several decades. In particular, erbium oxide coatings recently showed excellent permeation reduction efficiency. Hydrogen solubility in the coatings can largely influence on the permeability and tritium inventory in blanket systems. Nevertheless, the number of studies reporting hydrogen states in the coatings is limited. In this research, hydrogen concentration in erbium oxide coatings after permeation experiments was measured using nuclear reaction analysis (NRA) with a Tandem accelerator. Hydrogen distribution in the coating will provide important information for further clarification of tritium permeation mechanism. The erbium oxide coatings were fabricated by filtered-arc deposition and dip-coating technique on reduced activation ferritic/martensitic steel F82H substrates. The permeation experiments were performed with hydrogen or deuterium at 773–973 K. To retain the hydrogen distribution, the samples were rapidly cooled with keeping the partial pressure of hydrogen isotopes at the high pressure side after the permeation test. Subsequently, the concentration of hydrogen or deuterium was measured by NRA via the reactions of H(15N,alpha gamma)12C for hydrogen and D(3He,p)4He for deuterium. The coatings fabricated by filtered-arc deposition showed that the deuterium concentration was several hundred atomic ppm at the surface, and gradually decreased with depth from the coating surface toward the substrate after introducing deuterium at 973 K. On the other hand, the concentration after introducing at 773 K increased from the surface to the substrate. In the presentation, we will discuss the permeation mechanism including the results of hydrogen permeation and concentration in dip-coated samples.

P2-046 **Neutronic assessment of a dual-coolant blanket design for DEMO**

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The purpose of this communication is to present the last results of a dual-coolant blanket design for DEMO developed within the framework of the Spanish Breeding Blanket Technology Programme TECNO_FUS, which objective is to explore the engineering functionalities of Dual-Coolant He/Pb15.7Li blanket for DEMO reactor design. The geometrical and materials specifications established in a previous work after an optimization process performed over a simplified 3D model, have been applied to a more detailed design, modifying material and dimension of Blanket, Shield and Vacuum Vessel in order to fulfill the priority requirements of a fusion reactor. The fundamental neutronic responses of this detailed and more advanced design has been evaluated. Tritium breeding ratio, power amplification factor, deposited power in the TF coils and helium production in steel components, have been calculated in order to demonstrate the functionality and viability of the reactor design and the fulfillment of basic capabilities such as to guarantee tritium self sufficiency, power efficiency, plasma confinement and reweldability of the lifetime components made of steel.

P2-047 **Behaviour of the Pb-Li Eutectic Alloy Impurities by ICP-MS**

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The liquid metal LiPb blanket design is one of the most promising designs for future fusion power reactors and under wide research in the world. Given that any variation in the alloy composition can generate important consequences in their behaviour and therefore envelope the performance of its regenerating function, a complete characterization of the Li-Pb eutectic alloy is required on the fusion technology field.

There are no impurities content studies of any materials produced by METAUX-SPECIAUX, Jost-Hinrich Stachov Metalhandel and the Institute of Physics of the University of Lithuania (IPUL), related to the Pb-Li eutectic alloy. One of the first documents on the fixing requirements regarding a maximum value of impurities in the production of Pb-Li eutectic is EFDA document 05/998 'Procurement of the Pb-Li eutectic alloy for EBBTF facility and for dedicated neutronics experiment Annex A. Technical Specification '.

One of the most critical impurities is Bi. The presence of Bi in the alloy causes the production of radioactive 210Po as consequence of the Bi neutron activation. Its content should be kept below tens of ppb (ng.g-1).

The aim of this work has been to optimize the quantification and establish detection limits of 8 elements (Al, Ti, Cu, Zn, U, Bi, Ag, Tl) present in the Pb-Li eutectic alloy by Inductively coupled plasma mass spectrometry (ICP-MS). It also has been studied some of these elements behaviour with respect to the major components of the alloy.

The experimental procedure was carried out sampling different regions of the ingots and the subsequent analysis by ICP-MS with collision cell (CCT). As a previous step to the instrumental analysis, it was necessary to establish an acid digestion method. The next step was to propose a standard procedure that allows a comprehensive quality control of both the main content of the eutectic alloy, as the maximum permitted levels of trace elements (impurities considered for this material).

P2-048 **Testing of Porous SiC With Dense Coating Under Relevant Conditions for Flow Channel Insert Application**

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The insulating Flow Channel Insert (FCI) is a critical component of dual-coolant blanket concepts of DEMO that provides thermal and electrical insulation between the RAFM structural steel and the eutectic liquid Pb-15.7Li cooling alloy, and minimizes MHD pressure loss. FCIs must be inert in contact with Pb-15.7Li and show low tritium permeability. In addition, FCIs have to exhibit sufficient mechanical strength to withstand strong thermal gradients and thermo-electrical stresses during operation. Silicon Carbide (SiC) materials fulfil the operational requirements for FCIs. Besides, porous SiC is an attractive candidate thanks to its relatively low cost manufacturing route to obtain a thermally and electrically insulating structure. To prevent tritium permeation and corrosion by Pb-15.7Li a dense coating shall be applied on the porous SiC. This work summarizes the results obtained within the program CONSOLIDER INGENIO 2010 for Fusion Technology TECNO_FUS on the development and testing of coated porous SiC for FCIs. Porous SiC was obtained following the powder metallurgy route and the sacrificial template technique using Al2O3 and Y2O3 as sintering additives and a carbonaceous phase as pore former. Sintering was performed in inert gas at 1850-1950ºC during 15 minutes to 3 hours, followed by oxidation at 650ºC to eliminate the carbonaceous phase and obtain the desired porosity distribution. The most promising bulk materials were coated by CVD with a ~15 microns SiC dense coating. Results on porosity, bending tests, thermal and electrical conductivity are presented. In addition, the microstructure of the coating and its adhesion to the porous SiC is studied. First results on hydrogen permeation in SiC and its corrosion under Pb-15.7Li are also shown. Finally, a correlation of properties with microstructural features is also discussed. It has been found that it is possible to obtain a dense coated porous SiC material with tailored thermal and electrical conductivity and mechanical strength by varying the manufacturing parameters.

P2-050 **Interaction of titanium beryllide with steam at high temparatures**

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Metallic beryllium has been considered as a leading candidate for a neutron multiplier in water- or helium-cooled solid breeder blankets in future fusion reactors. However, beryllium is highly reactive with water vapor at temperatures above 873 K, producing hydrogen that may lead to accidents due to H2 gas explosion. Titanium beryllides such as Be12Ti have many advantages over beryllium from the perspectives of higher melting point, lower chemical reactivity, lower swelling, etc. Therefore, Be12Ti material has attracted attention as an alternative of beryllium, and could be used as an advanced neutron multiplier in the fusion reactor blankets. Nakamichi et al. have developed new synthesis process of Be12Ti material called â€œplasma-sintering methodâ€. This process consists of loading raw material powder in the punch and the die unit, direct current pulse plasma generation to activate the surface of powder particles, and uniaxial pressing to enhance the sinterability. We had intensively studied H2 generation caused by interaction of water vapor with the titanium beryllide sample under a temperature-programed condition. In this study, we investigated interaction of water vapor with titanium beryllide under different temperature conditions such as stepwise temperature increase or constant temperatures. The results showed considerably different oxidation behavior of titanium beryllide. In the presentation, more details of these experimental results are discussed.

P2-051 **Determination of the protective atmosphere during the Pb-Li alloy fusion. Part I. Furnace atmosphere and temperature.**

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The nuclear fusion technology planned as future massive source of energy. Focusing on the ITER project, the heat energy generated in the reaction may heat the materials that involve the reactor and it is necessary cooling this materials. For this reason, one of the most critical parts of its design are regenerating wrappers or blanket modules. Between the different materials proposed for this purpose we have been studied the Pb-Li eutectics alloy. For the production of the eutectic alloy is of vital importance to ensure efficient mixing of raw materials, hampered by the large difference in density of the two components and the high reactivity of Li versus O2. Furthermore, it is necessary to minimize the presence of intermetallic aggregates, owing to their high melting temperatures. In this work, different tests were made with eutectic ingots in order to determine the composition of the protective atmospheres and the crucible material. The alloys manufactured were characterised by microscopy and ICP-MS. The results demonstrate that the crucible material is not one of the most influential variables, anyway, the best choice when making mergers is the SiC, due to the low reactivity presented with Pb-Li. Regarding the protective atmosphere, ArBIP showed the best protection and can be used in a wide range of temperatures. The experimental procedure performed use a LiCl-KCl salt as slag during the melting process. The slag formed protects the alloy against oxidation and removes the impurities. In this paper the atmosphere optimal conditions for melting ingots process are presented as follow:

•N2(TEC) to temperatures of 600°C, since at higher temperatures is a higher oxidation of lead is produced

•ArBIP to temperatures of 800°C. This gas has very good properties for the protection of the alloys, since it can be used both at low and high temperatures.

P2-052 **Definition of the key parameter for production of eutectic Pb-Li alloys. Part II. Mass balance and temperature.**

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The development of thermonuclear experimental reactor (ITER), whose objective is to produce energy, has raised the study of Pb-Li eutectic alloy, as they has been selected for the manufacture of HCLL TBM.

However, during the manufacturing process of the Pb-Li alloys, there is a loss of Li which inhibits the formation of eutectic structures. In this work, we have performed the fusion process from pure Pb and Li to obtain Pb-17%at.Li (eutectic alloy), evaluating different parameters that control the process. Therefore, it has been necessary to desing a furnace for the production of these alloys based on variables process such as inert gas, melting temperature, crucible material and the possible use of fluxes (LiCl-KCl). Modifying the protector gas and the %wt. of fluxes, the lost of Li was minimized during the melting. Thereafter, we have done the fusion process in different furnaces, using the most favorable conditions obtained in previous studies. This study demonstrates that one of the most significant variable is the design of the distribution of the starting materials in conjuction with others as the temperature and isotherm. Cast alloys performed were characterized both analytical and microstructural form, wich implies the determination of the compositional distribution of Li in the ingot, the phases present, and the microsegregation present. The paper presents the optimal conditions for melting process that have been obtained for the production of Pb-Li eutectic alloys, based on the use of LiCl-30%wt.KCl in the presence of Ar BIP at 450°C during 5 min. It happens because when temperatures are lowers a homogeneous melt cannot be assured due to the melting point of lead is around 327°C and the eutectic salts is at 350°C. It is very important to set the initial %at. Li around 4 over the one marked by the eutectic alloy (Pb-16%at.Li).

P2-053 **Effect of post-weld heat treatment on irradiation hardening of the weld metal of low activation vanadium alloys**

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Welding is an essential technology for the low activation vanadium alloys for application to fusion blankets. However, the weld metal can become more brittle than the base metal by neutron irradiation. Post-weld heat treatment (PWHT) is known to improve the brittle characteristic of the weld joint. The purpose of this study is to estimate PWHT effects on hardness, microstructure and impact properties of the weld after irradiation. Bead-on-plate welds were made from the reference V-4Cr-4Ti alloy, NIFS-HEAT-2, by electron beam welding. PWHT was conducted at 673-1273K for 1h. Irradiation with 11MeV H+ was performed at 573K for 23hrs in the FFAG accelerator in KURRI. Hardness around the weld bead was measured by Vickers hardness tests and nano-indentation tests. Charpy impact tests were conducted at 77-283K. Microstructural observations were carried out with transmission electron microscope (TEM). The weld metal was harder than the base metal in as-weld condition. Hardness of the base metal was increased slightly but reduced to the level similar to that before welding after PWHT at 673K and higher. On the contrary, the weld metal exhibited additional hardening by PWHT at 873K. Hardness of the weld metal was recovered to the level before welding after PWHT at 1173K. The TEM analysis clarified that the Ti-CON precipitates were dissolved during the welding. Therefore, interstitial impurities were expected to cause solid solution hardening by the welding. The hardening by PWHT at 873K is due to formation of submicroscopic precipitations, according to previous studies. The precipitates were inhomogeneous and formed islands by PWHT at and above 1023K. Hardness of the base metal before and after the irradiation was 130Hv and 170Hv, respectively, so that irradiation hardening was 40Hv. Effect of PWHT on irradiation hardening of the weld metal and the results of Charpy impact tests will be presented and discussed.

P2-054 **Loop heat pipes for energy conversion in fusion reactors**

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Fusion-to-thermal energy conversion systems of magnetically and inertially confined plasma reactors are today in an early stage of development and require significant improvements in simplicity, reliability, energy conversion efficiency, safety, and ability to handle steady-state and plasma disruption loads. Loop heat pipes are heat transport systems that could fulfill these requirements. The operation of a loop heat pipe is based on the evaporation and condensation of a working fluid to transfer heat and the capillary forces developed in the porous wick of the evaporator to circulate the fluid through the loop. The evaporators and condensers of loop heat pipes can be physically removed from each other, which allows for an effective design that employs liquid metal and molten salt coolants that can also serve the function as tritium breeders. Loop heat pipes can be arranged with multiple evaporators, condensers, and compensation chambers, and could be of modular design that allows easy replacements of modules, for either repair or removal to recover tritium.

This paper presents results from a thermo-hydraulic analysis of a prototype liquid metal-cooled loop heat pipe system with one or more evaporators and condensers that fits within the ITER's port for testing the test blanket modules. The modeled prototype accounts for the anticipated fusion reactor heat and neutron fluxes, single and two-phase flows in different components of the loop (liquid and vapor lines, evaporators, condensers, compensation chambers), and for different operating conditions of the loop that depend on the components design and imposed heating fluxes. The system's startup and its operation in a variable and fixed conductance modes is also addressed.

P2-055 **First Wall Assembly Technologies of 1/3 Scale China Test Blanket Module**

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The dual functional lithium lead - test blanket module (DFLL-TBM) is chosen as one of the candidates for China TBM for its advantages of tritium breeding and excellent heat exchange performance, and China low activation martensitic (CLAM) steel as its candidate material. The fabrication technologies of the first wall (FW), one of the most important components, was studied and presented. Firstly, the CLAM rectangular tubes were drawn with round tubes, and the mechanical properties and dimensional accuracy of CLAM rectangular tubes were tested. After that the rectangular tubes and plates were assembled with hot isostatic pressing (HIP) - diffusion welding (DW). The FW of 1/3 scale ITER TBM was fabricated with electron beam welding (EBW) and HIP - DW, and the EBW was a pre-processing of HIP - DW for pre-sealing of the joint. Finally, the reliability of the FW mockup was evaluated with non-destructive testing under 8 MPa water flow.

P2-056 **Neutronics analysis of the shielding performance of HCCB TBM**

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The helium cooled ceramic breeder (HCCB) test blanket module (TBM) will be installed in the equatorial port of ITER to perform tritium breeding blanket related test. TBM system should provide efficient shield for TF coils with low nuclear heating during operation and for human access to the areas behind TBM module for maintenance and replacement during shutdown. The shielding performance of HCCB TBM was studied. Neutron and gamma transport was simulated using MCNP code and FENDL/MC-2.1 data. Activation calculation was performed with FISPACT-2007 code and EAF-2007 activation file. In this study, shielding block was optimized according to a mass limit. The values of nuclear heating deposited in TF coils were analyzed for different cases. The nuclear heating in TF coils decreases with high content of water in shielding block. The dominant nuclides of activity and dose rate in shielding block were 51Cr, 55Fe, 60Co, 58Co, 182Ta, respectively. The shutdown dose rate after 12 days cooling in pipe forest area were obtained and discussed.

P2-057 **Design Progress and Performance evaluation of a Korean HCCR TBM in ITER**

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Korea has designed a Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM) to be tested in ITER. It has four sub-module considering the delivery for Post Irradiation Examination to home country and fabricability, and the breeding zone consists of Be multiplier, Li ceramic breeder, and graphite reflector. The design of the main structural components such as First Wall (FW) and Side Wall (SW) was obtained through the optimization process to meet the design requirement. The breeding layer thicknesses were obtained through optimization considering neutronic and thermal hydraulic aspect. For coolant and He purge gas distribution in breeding zone, flow manifold and scheme were suggested and evaluated with the CFD code.

P2-058 **Numerical Analysis of the Heat Transfer Coefficient for Blanket Shield Block by Simplified 3D Model**

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The blanket system consists of 440 Blanket Modules (BMs) with an 18-fold segmentation in the poloidal direction. Each module is composed of a First Wall (FW) and a Shield Block (SB). The main function of the SB is to provide nuclear shielding and supply the FW panel with cooling water. In the conceptual design activities, a poloidal cooling concept of SB was proposed and its feasibility was verified. In the preliminary design, the hybrid cooling concept, which consists of poloidal and radial cooling concept, was adopted to enhance the cooling capability. In the design activities, the low coolant velocity was observed in some parts of radial cooling and cooling header from the thermal-hydraulic analysis. According to definition of Heat Transfer Coefficient (HTC), it is depend on the heat flow, heat transfer surface area and difference in temperature between the solid surface and surrounding coolant area. Also, the dependence of HTC can be represented by using dimensionless parameter; Nusselt number (Nu), Reynolds number (Re), Prandtl number (Pr), i.e. it is varied with thermal and flow characteristics of coolant. The objective of this study is to investigate the effect of cooling parameters on the HTC. The modeling and analysis of simulated model was implemented by using the ANSYS and CFX. Firstly, in order to investigate the effect of thermal characteristic of coolant, two types of coolants with different velocity and temperature were prepared. Secondly, the effect of geometrical parameters on the heat transfer coefficient was investigated.

P2-059 **Manufacturing and Testing of Full Scale Prototype for ITER Blanket Shield Block**

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Based on the preliminary design of ITER blanket shield block (SB) 08, the full scale prototype (FSP) has been manufactured and tested in accordance with pre-qualification program, and related R&D was performed to resolve the technical issues of fabrication. The objective of the SB pre-qualification program is to demonstrate the acceptable manufacturing quality by successfully passing the formal test program. 316L(N)-IG stainless steel forging blocks with 1.80L x 1.12W x 0.43t (m) were developed by using an electric arc furnace, and s a result, the material properties were satisfied with ITER specification. In the course of applying conventional fabrication techniques such as cutting, milling, drilling and welding of the forged stainless steel block for the manufacturing of SB 08 FSP, several technical problems have been issued. The machining for intermodular keyway and contact surface of key pad needs special angle head/tools and machining processes due to its complicated shape from interface cut-outs and hard-to-cut material. And also, the hydraulic connector of cross-forged material re-melted by electro slag or vacuum arc requires the application of advanced joining techniques such as automatic in-bore TIG and friction welding. Many technical issues - drilling, welding, slitting, non-destructive test and so on - have been raised during manufacturing. Associated R&D including the computational simulation and coupon testing has been done in collaboration with relevant industries in order to resolve these engineering issues. This paper provides technical key issues and their possible resolutions addressed during the manufacture and formal test of SB 08 FSP, and related R&D.

P2-060 **Numerical modelling of a tapered Holweck vacuum pump via linear kinetic theory**

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The Holweck molecular drag pump is widely used in the vacuum pumping industry. It can be a self standing apparatus or it can be part of a more advanced pumping system. It is composed by an inner rotating cylinder (rotor) and an outer stationary grooved cylinder (stator).

Vacuum pumps may be simulated by the DSMC method but due to the involved high computational cost, manufacturers commonly resort to empirical formulas and experimental data. Recently, a numerical simulation of a Holweck pump via linear kinetic theory has been proposed [1]. Neglecting curvature and end effects the gas flow configuration through the helicoidal channels is decomposed into four basic flows. They correspond to pressure and boundary driven flows through a grooved channel and through a long channel with a “T” shape cross section. The approach is computationally efficient but it is developed for pumps with channels of constant height from the inlet to the outlet.

In the present work this methodology is extended to tapered Holweck pumps (the channel height varies between inlet and outlet). A Holweck pump with tapered pumping channels has been solved by implementing the Navier Stokes equations subject to slip and jump boundary conditions [2] and therefore is valid only in the viscous and probably in the early transition regime. Here, the four basic flows, described in [1] are solved numerically based on the linearized BGK model equation to create a kinetic database of the flow characteristics for a large spectrum of the Knudsen number and various geometrical configurations. Based on this database and a methodology for computing the conductance through channels of variable cross section subject to arbitrary pressure drops, the performance characteristics of the Holweck pump are obtained. In particular, the throughput, the pressure ratio and the efficiency are computed according to the operational data such as the rotational speed and optimized in terms of the geometric data such as the number and the angle of the grooves. A comparison with available experimental data is performed demonstrating the effectiveness of the methodology.

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P2-061 **Design and development of hydrogen isotope sensor in liquid Pb-Li**

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Indian LLCB – TBM uses liquid Lead-Lithium (Pb-Li) as tritium breeder, neutron multiplier and coolant. Tritium bred in liquid PbLi stream has to be recovered by Tritium Extraction System. Therefore, a reliable sensor with quick response time for measurement of hydrogen isotope is essential. A hydrogen isotope sensor in liquid Pb-Li, based on permeation of hydrogen isotopes through metal (sensor material) is designed. The capsule shaped sensor, made of iron membrane coated with Pd from inside (downstream side), allow hydrogen isotope to permeate through it. The design work mainly includes the selection of proper thickness which is supported by numerical calculations for optimization of maximum permeation flux and fast response. The numerical calculation utilizes a physical model having recombination of two hydrogen isotope atoms at the surface and atomic diffusion through the bulk. In this work, design calculations, fabrication procedures and characterization of the hydrogen isotope sensor will be presented.

P2-062 **Design and R&D Activities of TriPla-CA Consortium in support of ITER Tritium Plant development**

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The design of ITER tritium processing systems shall take benefits of experimental data and process validation on experimental facilities that are ITER relevant. TriPla-CA, which is a consortium of several EU laboratories aiming to support tritium relevant activities for ITER design is carrying out R&D and design for ITER Tritium Plant. Several rigs and experimental facilities have been enhanced and developed in order to explore a wide range of envisaged scenarios of some ITER tritium plant systems such as Water Detritiation System (WDS), Isotope Separation System, highly tritiated water processing and atmosphere detritiation. Beside the experimental activities the enhancement of the relevant software for simulation and design of various tritium processing systems is ongoing. The main achievements concerning the R&D, the mechanical design including the seismic calculations of some ITER WDS components and the main expertize and the hardware available inside the consortium will be presented.

P2-064 **DINS method for diagnosing fuel retention in plasma facing components**

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Tritium retention into the ITER plasma-facing components is a subject of increasing concern. In this project we propose to validate the Deep Inelastic Neutron Scattering (DINS) method for non-destructive, bulk diagnosis of fuel retention in wall tiles. DINS [1] is based on the scattering of epithermal neutrons (typical energy 1 eV < En < 50 eV) on a sample: scattering occurs with the nuclei with typical exchanged energy and momentum of 1 eV to 30 eV and 30 Å-1 to 200 Å-1, respectively. In these ranges, the scattering can be considered as performed on an almost-free particle. The scattering spectrum clearly separates the peaks due to scattering from Hydrogen, Deuterium and Tritium (due to their mass ratio 1 to 2 to 3), while peaks due to scattering on other nuclei is superimposed. The area over the peaks due to these three isotopes is proportional to the total amount of them. The consequent estimation of total amount of hydrogen isotopes in the sample is thus straightforward (even if background subtraction and peak area fitting have to be carefully assessed [2]). DINS experiments can be performed at the VESUVIO beamline at the ISIS neutron source. VESUVIO [3] is a recently-improved inelastic neutron spectrometer specialized for DINS: its under-moderated beam provides a high flux of neutrons in the interesting region. The DINS spectra are recorded through the time-of-flight (ToF) technique. In this contribution, we report about preliminary calculations and simulations performed with the DINSMS package, necessary for the validation of the technique. Comparison with experiment will be discussed.

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P2-066 **Design Concept of the ITER SDS Getter Bed**

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A key design objective for the ITER Storage and Delivery System (SDS) is to develop a metal hydride bed that meets the unprecedented levels of performance required for fuelling of the Tokamak and which optimises the design against needs which can be conflicting, while at the same time respecting safety requirements for the confinement of tritium. The hydride material of choice for storing the deuterium and tritium fuelling gases is depleted uranium (DU). A foremost requirement is the reliable confinement of tritium in all normal, incidental and accidental situations. This need affects not only the structural requirements of the design, but also the selection of particulate filters to ensure unacceptable levels of particulates do not exit from the hydride bed. Compatibility with the ITER authorized domain, as represented by the preliminary safety report (RPrS), will be ensured, in particular with respect to waste management. The performance requirements for the bed are obtained from the fuelling needs of the ITER machine. The ability to deliver gas at a high rate and to quickly reload with gas recycled via the isotope separation system place a need for very effective heat transfer for heating and cooling of the hydride material. A separate need to establish and maintain precise static and isothermal temperatures for tritium inventory measurements by calorimetry mean that high levels of thermal isolation are also needed at such times. This ability will be met by locating the storage vessel for the hydride material within an outer jacket which can be gas filled or under vacuum, depending upon the operations being performed by the hydride bed. The paper will describe how the multifarious functional, engineering and performance requirements for the ITER hydride bed will be brought together into an optimized design. The paper will also give the current status of a full-sized prototype bed design.

P2-067 **Direct Measurement of Tritium Production Rate in LiPb with removed parasitic activities: preliminary experiments.**

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In the Helium Cooled Lithium Lead (HCLL) concept applied to the Test Blanket Modules (TBM), Tritium Production Rate (TPR) is the basic parameter, which needs validation in experimental way. Liquid Scintillation technique applied for direct measurement of Tritium activity produced in LiPb eutectic in Frascati HCLL TBM mock-up neutronic experiment has been tested so far in the case of LS measurement after long period since irradiation. Calculated with the use of FISPACT parasitic activities, mainly isotopes of lead (209Pb, 204mPb, 203Pb), and 76As 1.4 hour after irradiation, exceed ca 5 times Tritium activity making direct measurement very difficult to perform. To avoid delay of some weeks required for "cooling" the irradiated samples, we propose removal parasitic isotopes from the measured solution in chemical way. Samples (1 gram of LiPb eutectic) irradiated in reactor fast neutron flux (fluency 2.53E18 cm-2) were diluted in acid solution and metallic cations removed by precipitation using in test experiments 4 different reagents: potassium iodide (KJ), strontium chloride (SrCl2), APDC (C5H8NS2·NH4), and PAN. Precipitation procedure in each case lasted ca 5 min., and the following filtration next some minutes. Each filtrate (ca 120 ml) was measured with the use of HPGe detector and in TXRF analyzer mainly with respect of Pb concentration. Obtained results of Pb removal by SrCl2 followed by PAN lower parasitic Pb radioisotopes to ca 8 ppm what corresponds to less than 1% of initial Tritium activity (1.4 hour after irradiation). Remarkable input coming from 76As and also from other less abundant isotopes is easy to avoid by usage of high purity LiPb. Specific Tritium activity generated in LiPb sample and determined in LS spectrometer was 41.5 ± 0.8 kBq/gLiPb and Tritium activity of filtrates showed no dependence upon used reagent being within measurement error the same as for analyzed LiPb samples.

P2-068 **R&D on Hydrogen in the Liquid Lithium Loop for TechnoFusion**

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The success of nuclear fusion as a commercial energy source requires the development of several technologies related to long-term research projects. Among the nuclear fusion challenges, liquid metal technology is one of the key aspects that need to be developed. In particular, liquid lithium is an essential element for tritium breeding and, in some designs, it is also proposed as coolant. Nowadays there is an important shortage of experimental facilities devoted to the R&D related to these issues. Among the facilities in TechnoFusion, a liquid lithium laboratory is planned to be constructed. Initially, it will be devoted to the understanding of liquid metal corrosion phenomena and the interaction between liquid lithium and structural materials. On a second phase, the research topic will focus on the study of the solubility, diffusivity, retention and permeability of tritium (hydrogen). Indeed, mastering the tritium recovery will greatly increase the economic feasibility of commercial fusion energy and its fuel cycle. This contribution summarizes the basic layout of the TechnoFusion lithium loop and the definition of some experiments related with the recovery of tritium (hydrogen) from lithium.

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P2-069 **Separation of fuel and Impurity particles using divertor simulator TPD-Sheet IV**

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The pumping of helium ash has become important for the control in the edge plasma of the divertor because of the helium ash has limit of concentration. The methods to improve the helium removal performance of a pump limiter by using a RF ponderomotive force (RF-filter) were presented. The selective removal of helium ash using by ion cyclotron resonance (ICR) method has been studied in a linear divertor simulator, TPD-Sheet‡W[1]. We have demonstrated the ICR method of the helium or helium/hydrogen sheet plasma by the RF electrodes of two parallel plates, sandwiching the plasma. Measurements of the ion temperature in the plasma were carried out a fast scanning Faraday cup. In addition, the ion densities in the plasma were measured by an omegatron mass analyzer and the neutral densities of resonant ions was measured by a quadrupole mass analyzer. As a result, the ion densities of He+ decrease with increasing the RF power. It is found that the selective removal of the helium ions in the sheet plasma is successful by ICR method.

Topic D Materials

P2-070 **TEM Characterization of He Effects in First-Wall Structural Materials Under Fusion Relevant Conditions**

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Reduced activation ferritic/martensitic (RAF/M) steels and oxide dispersion strengthened (ODS) ferritic alloys are potential first-wall structural materials for fusion power systems, but their suitability for high-dose service remains to be established. Helium can lead to degradation of mechanical properties and dimensional instability.

A novel experimental technique was employed to study He at fusion relevant conditions. A thin layer of NiAl was deposited on the surface of a TEM disc, and He was produced when irradiated in the HFIR reactor via a two-step reaction. Helium was injected to a depth of a few microns, along with neutron-induced displacement damage.

We performed TEM image simulations of He bubbles to assess the relationship between the first dark Fresnel ring seen under defocused conditions and actual bubble size. Calculations show that He bubbles measured this way are consistently larger than the actual bubble when imaged with a highly incoherent electron beam, but just the opposite if the beam is coherent. The difference between measured and actual bubble size were found to increase with increasing defocus. Electron beam accelerating voltage, bubble size, bubble position, and sample thickness do not significantly affect the measurement.

We report initial measurements on two ODS alloys (PM2000 and 14YW) and one RAF/M steel (modified F82H) with a 4.7 µm thick NiAl layer that were irradiated to 21.2 dpa and 1230 appm He at 500°C. The PM2000 contains a high density of small (< 2 nm) He bubbles in the matrix, while at the interface of Y-rich particles large voids were found. Both 14YW and modified F82H exhibited a large number of voids. The voids in 14YW coexist with small Y2O3 particles typically less than 10 nm. Dislocation loops in all three samples were also quantified. A higher density of <100>{100} loops was found than that of 1⁄2<111>{111} loops.

P2-071 **Depth-dependent nanoindentation hardness of reduced-activation ferritic steels after MeV Fe-ion irradiation**

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The irradiation hardening behavior of reduced-activation ferritic steels after single Fe-ion beam irradiation and dual-ion (Fe ion and He ion) beam irradiation experiments was investigated with a nanoindentation test. The ion-irradiation experiments were conducted at 270 ‹C with 10.5 MeV Fe3+ ions up to 5 dpa at a 1000 nm depth from the irradiated surface. The materials used were F82H, F82H+1Ni, and F82H+2Ni to investigate the effect of Ni addition on the irradiation hardening behavior. The measured nanoindentation hardness was converted to the bulk-equivalent hardness based on a combination of the Nix-Gao model to explain the indentation size effect and the composite hardness model to explain the softer substrate effect of the non-irradiated region beyond the irradiated depth range. It is clearly shown that the Ni addition enhances the irradiation hardening of F82H. The bulk-equivalent hardness is compared with the experimentally-obtained Vickers hardness of F82H steels after neutron irradiation. The effect of simultaneously-implanted helium on the irradiation hardening is negligible in the investigated irradiation conditions.

P2-072 **Diffusion Bonding of 9Cr ODS Ferritic/martensitic Steel with the Phase Transformation**

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Diffusion bonding was employed to join oxide dispersion strengthened (ODS) ferritic/martensitic steel under uniaxial hydrostatic pressure using a high vacuum hot press, and the microstructure and tensile properties of the joints were investigated. 9Cr ODS steels were successfully diffusion bonded at 1150 oC for 1 h and followed heat treatment including a normalizing at 1050 oC and tempering at 800 oC for 1h. Diffusion bonding with phase transformation process resulted on a favorable microstructure without inclusion and micro-voids on the bonding interface or the degradation in the base metal. TEM observation revealed that the nano-oxide particles on the bonding interface were uniformly distributed in the matrix and that the chemical composition across the bonding interface was significantly constant. At room temperature, the joint had nearly the same tensile properties with that of base material. The tensile strength of the joint region at elevated temperatures was comparable with that of the base metal. The total elongation of the joint region slightly decreased but, it reaches 80% of base metal at 700 oC and the ductile fracture occurred where is far from the bonding interface. Therefore, it is considered that diffusion bonding with the phase transformation could be a very useful joining method to fabricate component in terms of the next generation nuclear systems with 9Cr ODS ferritic/martensitic steels.

P2-073 **Fatigue Property Change of Pure Tungsten Due to Heat Treatment**

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Tungsten is promising plasma facing material (PFM) of fusion reactor because of its high melting temperature, high thermal conductivity and high resistance to physical sputtering. Since the PFM must support dynamic loads induced by thermal and electromagnetic stresses, the fatigue property of tungsten must be understood. One of the major material issues of tungsten as a structural material is grain boundary embrittlement due to recrystallization. The recrystallization of the pure tungsten usually occurs due to the heat treatment above about 1300ºC. However, the effect of recrystallization on fatigue property of tungsten has not been well known. The objective of this study is to evaluate the fatigue property change of pure tungsten due to the heat treatment at the temperature ranges from 900ºC to 1300ºC, and to compare it with the other mechanical properties change such as the tensile property and hardness. The fatigue life of two kinds of pure tungsten (rod and plate materials) was evaluated at room temperature in air. As-received specimens of both two materials showed similar fatigue lives. Almost no change of the fatigue life was observed after the heat treatment at 900ºC. However, significant reduction of the fatigue life occurred due to the heat treatment at 1300ºC.

P2-074 **Selection of technology manufacturing ITER blanket modules flexible attachment from Ti-6Al-4V alloy**

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Ti-6Al-4V alloy was considered as a candidate material for flexible attachments of the shield blanket modules in the ITER reactor owing to advantageous combination of properties, i.e. low elasticity module, high resistance to impact loading, high strength, low expansion coefficient. The different technologies are used for manufacturing the flexible attachment made of this alloy. These are forging, stamping and pressing. Tensile properties, fracture toughness, low cycle fatigue and microstructure of forged and stamped Ti-6Al-4V alloy have been investigated in initial condition and after irradiation in the IVV-2M reactor at 26 0C to the dose of 0.1 – 0.4 dpa. In initial condition the strength and low cycle fatigue of both materials were practically the same, but the forged material had slightly better ductility and fracture toughness than stamped material. Irradiation resulted in hardening and significant decreasing of ductility and fracture toughness. The tensile properties under irradiation achieved in hardening saturation after a damage dose of about 0,2 dpa. The reduction of ductility and fracture toughness after irradiation were more essential for the stamped material. Minor deterioration of fatigue resistance due to irradiation has been observed in high strain range of both materials. Hydrogenation up to 700 – 800 ppm caused insignificant effect on ductility and fracture toughness of forged material, that let to consider the forging as preferable manufacturing option.

P2-075 **Numerical study of the flow conditioner for the IFMIF liquid lithium target**

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IFMIF (International Fusion Materials Irradiation Facility) is an accelerator based deuteron-lithium (D - Li) neutron source to simulate the neutron irradiation field in a fusion reactor. The target assembly of the IFMIF consists of the flow conditioners and the nozzle, which has to form a stable lithium jet. The free-surface stability of the high-speed lithium target of IFMIF is one of the crucial issues, since the spatio-temporal behaviour of the free-surface determines the neutron flux to be generated.

The stability of the flow upstream the nozzle exit is affected by fluid-dynamic perturbations caused by the piping elements, such as elbows, joints and valves. The use of flow conditioners aims to achieve a flat uniform velocity distribution with simultaneously low turbulence intensity. The present work is focused on the numerical study of the flow conditioner efficiency. The comparison of different types of flow conditioners is based on the detailed numerical analysis of specific hydraulic effects in pipe elbow and in flow conditioners. A number of flow conditioners such as tube bundles (honey comb), perforated plates (screens), rotation vanes and their combinations are investigated and analyzed. Calculations show that perforated plates are generally not effective with respect to a large scale swirl reduction. The honeycomb-screen combination improves the flow uniformity downstream, but increases the pressure drop. An incropartion of guide vanes upstream the elbow reduces most effectively the generation of secondary flow s and hence proves to be the superior option since it allows for a reduced conditioner length at simultaneously acceptable pressure loss.

P2-076 **Neutronic analysis for the IFMIF EVEDA Reference Test Cell and Test Facility**

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The IFMIF (International Fusion Material Irradiation Facility) project is in the so-called EVEDA phase aiming at producing a detailed and fully integrated engineering design under the framework of the Broader Approach activity. The Test Facility (TF) consists of several subsystems including the test cell (TC), which is the central part of IFMIF where an intensive neutron field is generated by d-Li nuclear reactions to irradiate candidate fusion reactor materials placed inside the test modules (TMs). During the EVEDA phase an optimized TC design has been proposed, developed and considered as the reference TC. In the present paper the details of the neutronic analyses to support the design work of TC as well as TF are described. A very detailed geometrical model for neutronic analyses has been prepared directly from engineering CAD data by utilizing the McCad conversion software developed at KIT. The geometrical model includes the detailed descriptions for the lithium target system proposed by Japan, three test modules, and the 3-dimesional arrangement of the biological shielding based on the reference TC design. The Monte Carlo code McDeLicious, which is an enhancement to MCNP5, has been utilized in order to adequately simulate the neutron and photon productions from the 6,7Li(d,xn) reactions in the lithium target. The present analysis is focusing on the following items which are important for the TF engineering design: the biological dose distribution around TC during operation, the nuclear heating distribution inside the biological shielding, and the nuclear property of the TC liner, in particular the He production rate that influences the possibility of welding in maintenance works of the liner. Some countermeasures for reducing the He production are also discussed.

P2-077 **Microstructural studies and surface analysis of laser irradiated in-situ Co–TiC composites.**

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Nowadays, investigation of different types of mixed materials samples are very important due to good understanding of erosion or the process of co-deposits' removal in terms of plasma-wall interactions in thermonuclear devices. The paper presents the results of investigation of the effects of the irradiation parameters on the microstructure and surface in-situ Co-TiC composites obtained by the high temperature solution method (HTS). The obtained ingots were irradiated with Nd:YAG laser pulses (Quantel Brio, 4ns, 70mJ, 1.06 μ m). The pulses were focused with a lens of the focal length of 10 cm. To investigate the dynamics of the crater formation process, in the consecutive irradiation, subsequent spots were irradiated by the number of shots increased by five, starting from one shot to fifty. Laser induced plasma was recorded by a quartz collimator connected to ESA4000 spectrometer with Echelle optics, in the spectral range from 200 to 780 nm. On the basis of observed Ti and C spectral lines, it was possible to estimate plasma parameters, namely electron temperature and density. Microstructure of the laser irradiated in-situ Co–TiC composites was characterized by scanning electron microscopy (SEM) and energy-dispersive X-ray spectroscopy (EDX). The Hirox digital microscopes systems have been used to 3D microscopic images analysis of the composite surface and structural changes after series of laser irradiation. Different kinds of mechanisms can play a role, depending on the type of material, laser irradiance, laser pulse length, etc. The study of melted crater ratio has been done on Co–TiC composites to investigate the effect of irradiation on composite surface microstructure. According to experimental results and analysis, it was shown that the observed craters on surfaces were deepened with increasing numbers of laser shot. On the other hand, heterogeneity of samples had influence on the depth and size of the crater.

P2-078 **Stress envelope of silicon carbide composites at elevated temperatures**

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A silicon carbide fiber-reinforced silicon carbide matrix composite (SiC/SiC composite) is a promising candidate material for the advanced fusion DEMO blanket because of the excellent themo-mechanical and chemical properties and irradiation tolerance of SiC itself. For the practical design of the DEMO, the stability of high-temperature strength of SiC/SiC composites needs to be identified. Additionally, strength anisotropy needs to be clarified because of its unique fabric architecture. This study therefore aims to evaluate mechanical properties by various modes such as tensile, compressive and in-plane shear at elevated temperatures, eventually providing a comprehensive stress envelope, i.e., strength anisotropy map, for the design. A plain-weave (P/W) Tyranno-SA3 fiber reinforced chemical vapor infiltration (CVI) SiC matrix composite was tested. Multilayered SiC/pyrolytic carbon (PyC) interface was formed on the fiber surface prior to matrix densification. Tensile and compressive tests were conducted by the small specimen test technique (SSTT) specifically arranged for the high-temperature use. In-plane shear properties were contrarily estimated by the off-axial tensile method assuming that the mixed mode failure criterion, i.e., Tsai-Wu criterion, is valid for composites. All tests were performed in vacuum. The preliminary test results in vacuum indicate no degradation of both proportional limit tensile stress (PLS) and the ultimate tensile strength (UTS) at temperatures below 800ºC. In contrast, slight degradation of UTS was identified at 1000ºC, although no significant degradation of PLS was expected. In this study, the key mechanism of such degradation will be discussed. Also, with high-temperature data of compressive and in-plane shear properties, the stress envelope at elevated temperatures will be provided for the design.

P2-079 **Fabrication and Characterization of Reference 9Cr and 12Cr-ODS Low Activation Ferritic/Martensitic Steels**

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Oxide Dispersion Strengthened Ferritic Steel (ODS Steel) is an attractive candidate for reinforcing fusion blanket components manufactured with Reduced Activation Ferritic/Martensitic Steels (RAFM Steels). There are a number of research going on for fabrication and characterization of ODS steels for fusion application. However, there is a lack of reference alloys available for various characterization efforts by a number of fusion research groups. This study aims at fabrication and basic characterization of reference ODS Steels by a similar fabrication processes for shared characterization by Japanese University research groups. For comparative studies of alloys with representative Cr levels, 9Cr and 12Cr-ODS, steels were produced. The fabrication proceeded with powder mixing, MA, encapsulation into mild steel cases, hot extrusion and hot forging at 1423K followed by final heat treatments (1323K for 1h and air cooling followed by 1073K for 1hr for 9Cr-ODS, and 1473K for 1hr for 12Cr-ODS). Each alloy was extruded into three bars. The characterization included chemical composition analysis, SEM and TEM microstructure, hardness and tensile test at RT and 973K. The chemical compositions of the resulting alloys are Fe-9.0Cr-0.14C-2.0W-0.23Ti-0.28Y-0.25O and Fe-11.7Cr-0.035C-1.9W-0.29Ti-0.18Y-0.084O. The estimated Y2O3 and excess O are 0.36 and 0.08% for 9Cr-ODS and 0.23 and 0.04% foe 12Cr-ODS, respectively. The Vickers Hardness was about 390 and about 320HV for 9Cr and 12Cr-ODS, respectively. 0.2% proof stress (YS) and ultimate tensile strength (UTS) were estimated to be 271 and 325 MPa and 251 and 281 MPa for 9Cr and 12Cr-ODS, respectively. For 12Cr-ODS the difference between UTS and YS is much smaller than that of 9Cr. The 9Cr-ODS was produced earlier and some research has already been carried out such as thermal creep test including characterization of aging effects, corrosion tests in Li-Pb or Li, and joining tests. Similar tests will begin soon for the 12Cr-ODS, which was produced more recently.

P2-080 **Effect of potential factors in manufacturing process on mechanical properties of F82H plate**

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Through the Broader Approach Activity, manufacturing technologies for the large scale melting of reduced activation ferritic/martensitic steel, F82H, have been studied and developed since 2007. F82H of over 3500 tons should be prepared to fabricate the blanket modules in a DEMO reactor. In order to prepare such large quantity of F82H with appropriate and stable mechanical properties, it is important to study effect of potential factors in the manufacturing process of mass production. In this work, effects of forging and cooling after normalizing, which have never been studied in the past researches on F82H, were focused. As for forging effect, plates with different forging levels were fabricated using different size slabs. Tensile and Charpy impact properties were then studied on these plates. No significant differences were observed in tensile property, but inhomogeneity and anisotropy were observed in Charpy impact property. From the result, it was revealed that forging should be performed enough to obtain an appropriate toughness. As for cooling effect, plates cooled in air and water after normalizing were fabricated. Tensile and Charpy impact properties were then studied on these plates. From the result, no significant differences were observed in the both properties. It was revealed that the cooling rate of air is enough to obtain appropriate mechanical properties.

P2-081 **Blanket material and technology developments toward DEMO under the Broader Approach framework**

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In the Broader Approach (BA) framework, research and development of the blanket related materials and technologies have been carried out in the EU and Japan. Those activities are implemented mainly at the Rokkasho BA site collaborating with universities in Japan. In the R&D on reduced activation ferritic/martensitic (RAFM) steels as the blanket structural material, 20-ton heat of the F82H RAFM steel has been successfully conducted by an electric furnace in 2012. Advanced neutron multiplier pebbles of beryllide (beryllium-titanium alloy) have been fabricated from the beryllide rod produced with the plasma sintering method. The reactivity of the beryllide pebbles with water is under investigation. Also advance tritium breeder of lithium-rich lithium titanate pebbles has been fabricated by an emulsion method, where the grain size of 2–3 μm is confirmed by the SEM observation. In the R&D on SiC/SiC composites for an alternative advanced structural material, mechanical properties of CVI- SiC/SiC plates have been obtained in high temperature vacuum environment up to 1000^{o}C. In the tritium technologies, corrosion tests of stainless steel with tritium water has started at the new tritium facility in the Rokkasho BA site. Based on the recommendation of the peer review working group consisting of the EU and Japanese experts for the BA DEMO R&D activities, following new four tasks have been initiated in 2013; 1) compatibility tests of SiC and SiC/SiC to LiPb for the flow channel insert application of SiC/SiC composites, 2) tritium recovery tests of advanced tritium breeder pebbles under 14 MeV neutron irradiation, 3) safety research on the plasma facing components of tungsten for ingress of coolant events, and 4) estimation of design windows related to material properties for DEMO blankets. Initial results of those new tasks will be reported on the conference.

P2-082 **Effect of Laser Beam Position on Mechanical Properties of F82H/SUS316L Butt-Joint Welded By Fiber Laser**

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A well perceived reduced activation ferritic/martensitic steel F82H is one of the most candidate structural materials for Test Blanket Module (TBM) for ITER because of its high temperature properties and weldability. In the current design of ITER-TBM, F82H pipe for water cooling has to be joined to SUS316L stainless steel pipe of ITER. Although some previous studies about the dissimilar joint between F82H and SUS316L using electron beam (EB) reported its applicability mechanically and metallographically, there is a strong demand to fabricate F82H/SUS316L dissimilar joint without a special environment such as a vacuum considering a practical construction process in the building site of ITER. Recently, as one of the high beam quality heat sources, a high-power fiber laser welding has been developed and the applicability of fiber laser for F82H has been reported, where argon or nitrogen gas was employed as shielding gas. Also, it has been reported that F82H and SUS316L could be butt-welded metallographically by using a fiber laser and controlling a laser beam position precisely. However, the mechanical properties of F82H/SUS316L butt-joints welded by a fiber laser have not been evaluated. So, in this study, F82H/SUS316L butt-joints were fabricated by varying the beam position of 4kW fiber laser precisely and their mechanical properties were evaluated through Charpy impact and tensile tests. According to a previous study about the weldability of F82H/SUS316L butt-joint, the beam position was shifted only to SUS316L side. In addition, the influence of post weld heat treatment (PWHT) was studied. From the serial experiments, it was found that the Charpy impact energies were increased by PWHT regardless of the laser beam position. Also, the fracture points under the tensile test were limited not to the welded metal but to the base metals. Moreover, the influence of welding speed on those properties will be discussed.

P2-083 **Overview of the Ceramic Breeder Materials Development**

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Lithium contained ceramics have been studied for fusion energy application for more than several decades. The world-wide efforts on ceramic breeder materials mainly focused on Li_2O, gamma $LiAlO_2$, Li_2ZrO_3, Li_4SiO_4 and Li_2TiO_3, with a clear emphasis on the development of Li_4SiO_4 and Li_2TiO_3 in recent years. Pebble fabrication processes have been developed up to 100 kg/a scale with no problem. In pile and out-of-pile test experiments have shown that the bred tritium can be extracted for different kind of breeder materials. Including experimental and computational study, the thermo-mechanical property of the pebble bed is under investigation for the lack of thermal property data in the TBM design activities. Several kinds of advanced tritium breeders were proposed to improve the thermal and irradiation property in Japan, EU and China. Including synthesizes and tritium release experiments, $Li_{2+x}TiO_{3+Y}$ were conducted to meet the DEMO demand in Japan. Li_4SiO_4+TiO_2 pebble was proposed to increase the mechanical property and Li_4SiO_4 pebble with several mole ratios of TiO_2 have been successfully fabricated in EU. Ceramic pebble within the Li-Si-Al-Ti-O system will be put into consideration serving as the advanced breeder materials towards CFETR, ITER and DEMO.

P2-084 **Influence of Cr Content On the Diffusive Transport Parameters and Trapping of Hydrogen in Fe Alloys**

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In this work, four alloys were tested by means of the gas evolution permeation technique in order to determine the influence of the Cr content on the hydrogen diffusive transport parameters: pure Fe metal, Fe5%Cr alloy, Fe10%Cr alloy and Fe14%Cr alloy. The experimental measurements were performed over the temperature range of 423-823 K and for high purity hydrogen loading pressures ranging from 1.0 • 104 Pa to 1.5 • 105 Pa. We observed that the permeability obtained for the pure Fe metal and for the two alloys containing 5 % and 10 % Cr follows an Arrhenius law in each case for this temperature range. For the alloy containing 14 % Cr, two different Arrhenius fittings were needed: one for high temperatures (above 710 K) and another one for low temperatures. In general terms, the increase in the Cr content in the alloy leads to smaller values of the permeability. Regarding diffusivity and Sieverts´ constant, trapping effects have been observed for the alloys containing Cr. These effects lead to a sudden decrease of the diffusivity together with an increase of the Sieverts´ constant at low temperatures. This phenomenon was detected at temperatures below 473 K, 523 K and 573 K for the Fe5%Cr alloy, Fe10%Cr alloy and Fe14%Cr alloy, respectively. According to the results, the influence of the metallurgical composition of Cr in Fe alloys on the transport parameters and trapping of hydrogen is discussed.

P2-085 **Annealing behavior of heavily neutron-irradiated beta-SiC on swelling and thermal diffusivity**

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Ceramic materials are expected to be used in several nuclear applications. In particular, very high radiation stability qualifies beta-SiC to be some of the most encouraging materials for thermonuclear fusion reactors. In these environments, ceramic materials suffer swelling and degradation of thermal diffusivity. Several study characterized the degradation changed with irradiation conditions, but the behavior of defects studied by measurement after annealing is not studied enough for heavily neutron-irradiated specimens.

In this work, measurements of macroscopic length changes (linear swelling) and thermal diffusivity change after isochronal annealing were carried out. Beta-SiC specimens were neutron-irradiated to 4-80dpa at temperatures of 646-1039K in the Japanese experimental fast reactor JOYO using the Core Material Irradiation Rigs, CMIR-4 and CMIR-5. Isochronal annealing was conducted every 100K up to 1773K in vacuum. Specimens were annealed for 1h at an objective temperature. After the annealing, specimens were cooled in the furnace, and each measurement was performed at room temperature. Thermal diffusivity was measured by the laser flash method with the t1/2 analysis using a specially ordered ULVAC TC-7000 that can measure ƒÓ3mmx0.5mm specimens.

Macroscopic length change was measured with conventional micrometer. Usually, swelling of SiC specimens were recovered linearly from the irradiation temperature, but in this study, heavily neutron-irradiated beta-SiC specimen recovered around 773K independently the irradiation temperature. For most specimen, the onset temperature was lower than the irradiation temperature, and the properties were not recovered linearly. Thermal diffusivity showed same tendency, and it showed these recovery were caused by recombination of frenkel defects. Both property changes showed transition around 1273K, that corresponds to the mobility of vacancy that cause void swelling.

P2-086 **First principles modelling of the initial steps of the ODS particle formation process in the alpha-Fe lattice.**

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Reduced activation ferritic-martensitic steels (RAFM) strengthened by yttria precipitates are promising structure materials for future fusion and advanced fission reactors. Oxide dispersion strengthened (ODS) particles hinder dislocation motion effectively resulting in higher strength and better high-temperature creep resistance of ODS steels in comparison to basic materials. Implementation of ODS materials widen the operating temperature as compared to conventional, RAFM steels as well as they are more radiation resistant. The size and spatial distribution of ODS nanoparticles significantly affect both mechanical properties and radiation resistance. Unfortunately the mechanism of the ODS particle formation is not completely understood. The very initial steps of the ODS particle formation process were modelled by first principles DFT/plane-wave method, as implemented in the computer code VASP. Stabilization of Y solute atoms in the alpha-Fe lattice was analysed in details. Binding energies for various combinations of Y solute atoms and vacancies were obtained for use in the future kinetic Monte Carlo calculations. Possible Y stabilization and precipitation reaction mechanisms were suggested. Y solute atoms create stable complexes with multiple vacancies. Stabilized by vacancies, Y solute atoms exhibit stronger attraction.

P2-087 **Observation of the Li target in the EVEDA Li Test Loop**

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The International Fusion Materials Irradiation Facility (IFMIF) is an accelerator-based D-Li neutron source aimed at producing an intense high energy neutron flux (2 MW/m2). To realize such a condition, two 40 MeV-deuteron beams with a total current of 250 mA are injected into a liquid Li stream (Li target) flowing at 15 m/s speed.

EVEDA (Engineering Validation and Engineering Design Activities) Lithium Test Loop (ELTL), which simulates hydraulic condition of the Li target and a purification system envisaged in the IFMIF, is a main Japanese activity of the Li target system in the IFMIF/EVEDA project. Construction and commissioning of the ELTL were completed in March 2011 in the Japan Atomic Energy Agency, and then the validation test has been performed since Sep. 2012. In the validation test phase, a series of tests on the Li target called the Li target operational tests were initially performed. The Li target in the ELTL is 25 mm in thickness and 100 mm in width, and flows along a concave back plate up to 20 m/s in velocity. The major test items and conclusions in the Li target operational tests are: 1) the start-up procedure of the Li target was examined and consequently no noticeable noise and vibration was confirmed; 2) the Li target up to the velocity of 20 m/s in a pressurized condition (0.12 MPa) and in a low vacuum condition was achieved stably and observed by imaging devices. Therefore, we completed preparation for detail tests on the Li target stability in which the thickness and surface fluctuation of the Li target are clarified by diagnostics such as a laser thickness meter and a high speed video camera.

Topic E Exvessel

P2-088 **Gas Exchange Processes on Stainless Steel Vessel Wall due to Interaction with Oxygen Contaminated Hydrogen Plasma.**

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The certain parts of stainless steel (SS) vessel wall of fusion devices are subjected to plasma irradiation. Oxygen is the most common impurity of plasma of today fusion devices and oxygen contamination is sure to be present in ITER plasma. This paper investigates the processes taking part during interaction of gas discharge deuterium plasma with oxygen addition with SS wall of the plasma chamber made from austenitic stainless steel containing 12%Cr, 18%Ni, 10%Ti. The range of oxygen concentration varied from 0.5 to 30%, and that of vessel temperature from 290 to 550 K. Irradiation of the vessel surface with ions or/and neutrals of (D2+O2) plasma in all investigated conditions led to number of effects like: activation of hydrogen diffusion from the vessel wall, practically total disappearance of oxygen concentration from the plasma, D2O, HDO, H2O, H2, and HD formation and emission from the wall surface, deuterium and oxygen trapping. TDS examination of the SS samples irradiated with (D2+O2) plasma showed that hydrogen release from SS was several times higher than deuterium trapping. The above mentioned processes are shown to be the result of the chain of the reactions initiated by interaction of deuterium atoms or ions with Cr2O3 molecule of the SS surface oxide layer. The dependence of the processes on oxygen concentration and vessel temperature are presented and discussed.

Conclusion is made that (D2+O2) plasma irradiation could be used for SS plasma vessel outgasing and for decrease of tritium inventory in vessel walls.

P2-089 **Ultrasonic Examination Feasibility Study for ITER Vacuum Vessel from Korea Domestic Agency**

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ITER vacuum vessel (VV) is a torus shape double walled box structure and it has a role to provide high vacuum and primary radioactivity confinement boundary. The vacuum vessel is categorized as safety important class (SIC) component class I and it shall satisfy RCC-MR codes (2007 edition) and the regulatory requirements such as French safety and quality order 1984 and French order of nuclear pressure equipment (ESPN).

According to these requirements, full penetration welding shall be applied into all of the VV welding lines and it requires 100% volumetric non destructive examination (NDE). Because the VV has lots of design interfaces with in-vessel components and ex-vessel components, the design is quite complex and there exist lots of welding lines with major wall thickness 60 mm. Many welding lines are unable to do the radiographic examination due to the design complexity, and hence, the ultrasonic examination is inevitable as an alternative NDE method. ITER grade stainless steel 316L(N)-IG materials shall be used for the VV and it makes worse to accomplish 100% ultrasonic examination because of high ultrasonic signal reduction at the weld areas.

For this reason, it has been studied the feasibility of 100% ultrasonic examination from Korea domestic agency (KO DA) and a main supplier Hyundai Heavy Industries (HHI) for the procurement of ITER VV sectors (two of total nine sectors) and ports (equatorial & lower ports). After the feasibility study, it was decided to fabricate three groups of ultrasonic examination blocks to develop details of examination methods and acceptance criteria take into accounting case studies for the ultrasonic equipments, ultrasonic probe types, probe scanning limitations due to the welding line geometries or obstacles, welding details, shape or configuration of artificial defects, and so on. Details of qualification test results for the 1st group of test blocks will be reported in this paper.

P2-090 **Thermocouple fixation and High Heat Flux Test on the ITER Neutral Beam Duct Liner mock-up**

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The ITER Neutral Beam (NB) duct liner is directly exposed to the high-power neutral beam during ITER operation. In order to withstand the high temperature heat flux from the neutral beam power, ITER NB duct liner which was made by CuCrZr panels should have intensive water cooling. Numerous thermocouples should be fixed on the NB duct liner surface for on-line monitoring and recording the temperature of ITER operation. In this study, engineering study of thermocouple fixation systems was performed to find out most reliable thermocouple fixation method. Twenty "K" type thermocouples with four different fixation systems had been embedded into NB duct liner full scale mock-up to measure the surface temperature of this mock-up during High Heat Flux Test (HHFT).

In order to recording the temperature precisely, Cu was electro-plated to the thermocouple end for each system. In addition to perform the HHFT, test jig had been fabricated considering the vacuum condition of HHFT. In order to simulate ITER operation during HHFT, test condition was determined by a number of discussion between ITER international organization (IO) and ITER Korea. Before the HHFT, thermal analysis had been performed to estimate the temperature and to evaluate the thermal structural reliability of NB duct liner mock-up during the HHFT. On account of the thermal loads applied on the analysis is not fully representative of the profile for power deposition, it was simplified with the thermal analysis beam direct interception profile + volumetric heating + HTC scheme. However, the volumetric heating has not been taken into account in the input given by IO for these actual HHFT tests. Finally HHFT had been carried out with thermocouples embedded NB duct liner full scale mock-up using electron beam irradiation facility of Korea Atomic Energy Research Institute (KAERI). The result of this test will be given the most reliable thermocouple fixation method and thermal distribution on the ITER NB duct liner during the HHFT which is simulates ITER operation.

P2-091 **Multi-Scenario Evaluation, Specification and Comparison of Electromagnetic Loads on ITER Vacuum Vessel**

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The ITER vacuum vessel (VV) is one of principal Safety Important Class components of the machine. The assessment of its structural performance requires comprehensive information about the loads to be available. The electro-magnetic (EM) transients cause mechanical forces, which represent one of the most critical loads for the VV structures, its supports and interfaces. The diversity of the loading factors and large number of analysis cases bring certain challenge to the assessment of the results of EM analyses and subsequent specification of the calculated loads for engineering purposes.

As a part of the continuous load specifications development and review process and in connection to the assessment of the overall balance of forces in the electro-magnetic system, the multi-scenario analysis and systematization of EM forces, acting on the ITER VV have been performed. The simplified mathematical model, together with corresponding computational technology, based on the use of integral parameters and operational analysis methods, provides a tool for the qualitative analysis of the problem in its breakdown and for effective computations and made possible the time-effective systematic assessment of a large number of scenarios. The developed model for numerical calculations includes a purely analytical modular processor for computing the field and mechanical loads. The applied functional approximation of data using methods of signal processing enables mesh-independent representation and specification of the results. The paper is focused on the results of analysis of EM forces acting on the ITER VV, including loads due to the toroidal eddy currents, poloidal eddy currents and halo currents, in toroidally-symmetric approximation like in the plasma MHD base model. The pressure load on shells and the net vertical forces, representing important parameters of the VV design and analysis, are addressed specifically. The results are compared with those obtained on large 3D FE models by transient analyses, and the essential agreement is found. The applicable approaches and principles of engineering specification of interactions, parameters, and necessary models are discussed. The obtained integral estimates exemplify the principal loads on the VV and can be used in specifications. The corresponding parameters and techniques can be used in verification and validation of the detailed complex models. The found envelopes and peak values in the envisaged scenarios, brokedown to separate components, can provide a database and reference for the load specification and analysis.

P2-092 **Design and fabrication of the ITER thermal shield**

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This paper describes the development status of the ITER Thermal Shield (TS) in particular the design, R&D and preparation for fabrication. To begin fabrication of the TS Main Components (TSMC) in 2013, the manufacturing design must be completed considering the manufacturing feasibility studies from industry and the latest interface development in the IO. This paper presents design details of the TSMC interfaces especially the physical interfaces of the supports and the clearances between the TS and adjacent systems. The TSMC are supported on the magnet and cryostat as follows: i) The inboard and outboard support of the Vacuum Vessel Thermal Shield (VVTS) are mounted on the TF coil structure, ii) The Support Thermal Shield (STS) is mounted on the magnet gravity support, iii) The Upper Cryostat Thermal Shield (UCTS) is supported on the cryostat top lid, and iv) The Lower Cryostat Thermal Shield (LCTS) is supported on the cryostat bottom floor. The TSMC will be cooled by gaseous helium (GHe) at 80K distributed by the TS Manifold (TSM). Structural and thermal behavior of the TS will be monitored by the TS Instrumentation (TSI). The final design of the TSM & TSI is to be completed in 2013. The design of the TSM & TSI and R&D results of the electrical breaker applied on the TSM are introduced in this paper. The electrical breaker R&D result shows that the design is robust. The Additional Neutron Shielding (ANS) tiles will be attached on the inboard area of the VVTS by bolting. To achieve the schedule milestone for fabrication of the ANS in 2014, the design shall be completed in 2013. In this paper, the design of ANS and the mechanical and physical properties of the candidate materials at room temperature and 80K are summarized.

P2-093 **Thermal-hydraulic analysis of ITER Vacuum Vessel Field Joints**

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The ITER vacuum vessel (VV) is located inside the cryostat and houses the in-vessel components. It is a double wall structure surrounding the plasma, where the volume between the external and internal shells is designed to allow the circulation of the cooling water through a very complicated structure of borated In-Wall Shielding (IWS), that work as neutron shield. The VV is made by 9 40° sectors, connected through splice plates to form the full torus: the regions at the interface between adjacent sectors are the so-called field joints. While each sector has its own cooling loop to remove the heat deposition due to nuclear heating, the cooling channels of the field joints are separated from the main vessel/port components. Individual inlet/outlet pipes for the water flow are thus provided for each field joint, located in the outboard bottom segment and on the upper port frame, respectively, to facilitate the leak-inspection procedures. The coolant flow splits in two streams, inboard and outboard respectively, passing through the borated IWS. Here we perform the thermal-hydraulic analysis of one field joint with the code FLUENT. We use two different levels of simplification: in the first one, all geometrical details of the IWS are retained; in the second one, the IWS is discretized in macro-structures. The water flow field, the temperature maps and the heat transfer coefficients are computed for both cases and compared in the paper.

P2-094 **Status of the ITER Vacuum Vessel Construction**

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The Vacuum Vessel (VV) of ITER has the essential function of being the first confinement barrier and removing nuclear heating during plasma operation. In order to fulfil its requirements, 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 accordance with French regulation Vacuum Vessel and ports are classified as Nuclear Pressure Equipment due to presents of radioactive products in the plasma chamber and in water cooled structure. The VV structurally supports all in-vessel components, such as blanket modules, divertor cassettes, port plugs, in-vessel coils and diagnostics, etc. It features ports at the upper, equatorial and lower levels, which provide penetrations for assembly and maintenance of the in-vessel components and host several systems and components.

Five Procurement Arrangements (PAs) with ITER Domestic Agencies (DAs) have been signed for the fabrication of nine sectors (seven sectors by the EU DA and two sectors by the KO DA), IWS (IN DA), upper ports (RF DA), and equatorial & lower ports (KO DA). The design of the VV system has been finalized and approved by an Agreed Notified Body (ANB). The KO sector fabrication has been approved and launched in February 2012. The EU sector fabrication is expected to be approved and launched by mid-2013.
All of the base materials, namely austenitic steel 316L(N)-IG as the main

vessel material, borated stainless steels and ferritic steel for the IWS to reduce neutron heating in magnets and to reduce magnetic ripples, have been specified to accomplish the system requirements. The on-site assembly scheme has been developed with the field joint welding of nine 40 degree sectors. Acceptance testing and criteria which include a pressure test, vacuum leak test, dimensional measurements, etc., have been specified. Maintenance of the VV, including In-service Inspection, is being developed to satisfy the safety requirements.

The EU DA has contracted the AMW Italian Consortium, formed by Ansaldo Nucleare, Mangiarotti and Walter Tosto as the main supplier for the seven European sectors and the splice plates joining the sectors. AMW has launched activities to prepare for the start of fabrication including work on the manufacturing design, material procurement, forming qualification, manufacturing documentation, construction of several mock-ups, welding procedure specification, NDE qualification activities, etc.

The KO DA has contracted Hyundai Heavy Industry (HHI) as the main supplier for two Korean sectors and equatorial & lower ports. HHI started fabrication of Sector #6 in February 2012 after approval of the fabrication documents and procedures by the IO and ANB. The RF DA has selected the Efremov Institute as the main supplier of the upper ports. The Efremov Institute has contracted MAN Turbodiesel AG, a German company, as the main sub-supplier. The IN DA has contracted with AVASARALA (India) for the IWS fabrication and pre-assembly. Base materials for the IWS are under procurement.

In this paper, the status of the ITER VV design finalization and fabrication will be described.

P2-095 **Mechanical Testing of the ITER Vacuum Vessel Support Structure - Coating Screening Tests and High Load Multi-Axial Mock Up Tests**

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The ITER Vacuum Vessel (VV) Support Structure is composed of nine hinges circumferentially distributed below each sector. These hinges have to comply with the VV-gravity and multi-axial disruptions loads in the MN range also allowing radial thermal expansion of VV during bake-out and operation. In order to avoid fretting and stick-slip effects during the radial motion, the dowels of the hinges are coated. A screening and qualification test campaign for selecting the optimal coating (MoS2, WS2) as well as the materials (steel and aluminium-bronze) has been performed under thermal vacuum conditions (200°C) and under a vertical load of up to 770kN with a modified test setup once used for the W-7 Narrow Support Element testing. For verification and testing of the mechanical performance of the complete hinge design, a scaled Mock Up (1/3 size) has been built. The dowels of the Mock-Up feature the selected MoS2 coating. For monitoring the mechanical behaviour, the Mock-Up is equipped with 73 strain gauges at locations given by a dedicated FEM analysis. The integration of the Mock-Up has been performed successfully, but also enabled manufacturing design optimizations. The mechanical tests are performed in two phases. Phase 1 was a one axis load test at 1,70MN to verify gravity load and radial expansion (simulated by rotation of the dowel itself). The results and inspection of the parts proved the mechanical stability of the system and also verified the positive effect of applying AlBz as counter material of the MoS2-coating on the steel dowels. Phase 2 is a two axis load test for verifying most significant Vertical Disruption Events loading. For this test, a special test rig is going to be developed with a capability of 5MN vertical loads and 2MN toroidal loads. The tests are scheduled to be finished before the ISFNT2013, and the test results will be presented.

P2-096 **Structural Analysis of the ITER Vacuum Vessel regarding 2012 ITER Project-Level Loads**

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A revision of the ITER Project-Level technical requirements related to load specification (to be used for all systems of the ITER machine) has been carried out in April 2012 for the following needs: (1) support ITER licensing by implementing requests from the French regulator; (2) include request from physicists for consistency with the plasma physics database and present understanding of plasma transients and electro-magnetic (EM) loads; (3) investigate the possibility of removing unnecessary conservatism in the load requirements and review the list and definition of incidental cases. Thus, the purpose of this study was to assess the impact of the new version of the ITER Project-Level Load Specification (LS) on the ITER Vacuum Vessel (VV) loads and the main structural margins required by the applicable French code RCC-MR.

The assessment proposed in this paper has taken place in two steps:

•The first one was to evaluate the update/additional events of the ITER Vacuum Vessel Load Specification according to the revised ITER Project-Level LS, such as category V events, category III seismic load (SMHV), category IV EM load type vertical displacement event, loss of coolant accident outside the cryostat (LOCA), loss of forced flow accidents in VV and in-vessel components (LOFA), load drop, etc.

• In the second step a preliminary assessment of the impact of these events has been performed on the structural margins calculated to guarantee the Vacuum Vessel structural integrity with regards to the RCC-MR code. The main Vessel and also several critical components such as Triangular Support, Blanket Module Attachment Keys, Gravity Supports and Ports have been considered and no relevant impacts have been found on the RCC-MR structural margins.

P2-097 **F4E strategy for the electromagnetic analysis of ITER components**

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Being based on a magnetic confinement concept, the ITER tokamak is a very complex electromagnetic system. In fact it is energized by a system of superconducting coils producing a peak magnetic field of 13 T and a stored magnetic energy bigger than 40 GJ, it is made of conductive, and in some cases ferromagnetic, structures on which instable plasmas with currents up to 15 MA can produce eddy and halo current circulation.

For all these reasons a careful electromagnetic assessment of the ITER device and of its main components is required for assuring reliable operations as well as their mechanical integrity.

In this context, Fusion for Energy (F4E), the European Domestic Agency for ITER, is involved in a relevant number of activities in the area of electromagnetic analysis in support of ITER general design, and of specific requirement of the EU in-kind procurement. In particular several ITER components (vacuum vessel, blanket shield modules and first wall panels, test blanket modules, ICRH antenna, in vessel viewing system, etc.) are being analyzed from the electromagnetic point of view.

In this paper we give an updated description of our main activities, highlighting the main assumptions, objectives, results and conclusions. The plasma instabilities we consider, typically disruptions and VDEs, can be both toroidally symmetric and asymmetric. This implies that, depending on the specific component and loading conditions, FE models we use span from a sector of 10 up to 360 deg of the ITER machine. The techniques for simulating the electromagnetic phenomena involved in a disruption and the postprocessing of the results to obtain the loads acting on the structures are described. Finally we summarize the typical loads applied to different components, give a critical view of the results and, when possible, we benchmark them against ITER specifications.

P2-098 **Structural Response of ITER Vacuum Vessel to Combustion Pressure Loads**

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Dynamic effects of internal pressure loads produced due to hydrogen combustion inside the vacuum vessel (VV) in case of LOVA accident may be very important with respect to the integrity of VV since it has relatively low design pressure 2.0 bar. Such a pressure could be easily exceeded depending on the scenario of a combustion process. The problem was that the design pressure is related to a static pressure load and this level could not be directly compared with a dynamic combustion pressure. In reality the mechanical response to dynamic pressure load has to be evaluated with respect to level of yield stress, will be or will not the elasticity limit exceeded. Structural response of the ITER vacuum vessel (VV) to combustion pressure loads was analytically evaluated for different combustion regimes. A simplified 1-D analytical model of the elastic oscillator for infinite cylindrical shell was used in current work. Real pressure profiles of the combustion process obtained from previous numerical simulations by GASFLOW and COM3D code in the cases of 4 and 20 kg of hydrogen inventory were used in a model in order to reproduce mechanical response of the VV structure to dynamic pressure loads, which reflects the different scenario of combustion process.

P2-099 **Thermo-structural optimization of the ITER ICRH Four Port Junction and Straps against in-vessel design criteria**

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The ITER Ion Cyclotron Resonance Heating (ICRH) antenna involves the launching and coupling of high power radio waves with plasma ions. The waves resonate with the gyration frequency of ions undergoing cyclotron motion, thus exciting them towards conditions required for ion fusion.

Within the framework of the CYCLE consortium agreement, design optimization work has been conducted on the ITER ICRH Four Port Junction (4PJ) and Straps assembly under the 'Normal Operation' conditions.

The aim of the analysis is to evaluate ways of making the component compliant with SDC-IC rules while balancing the competing demands of different performance requirements [1]. Of particular interest are the bends that connect the 316L(N) strap pipes to the housing, where previous work had shown that primary plus secondary stresses would result in a low predicted fatigue life. Coupled ANSYS CFX and structural models are used to simulate the effects of plasma radiation, RF losses, volumetric neutron heat loads and a 5MPa coolant pressure load.

Although all of the modifications explored resulted in primary plus secondary stresses exceeding the cyclic damage design criteria, some avenues are identified for future studies and a reduction in stress towards the target obtained.

[1] A. Borthwick et al.,2009, 'Mechanical design features and challenges for the ITER ICRH antenna', Fusion Engineering and Design, 84 (2-6), pp. 493-496

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Topic F Neutronics

P2-101 **ITER Components Cooling: Satisfying the Distinct Needs of Systems and Components**

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The ITER Tokamak requires multiple auxiliary systems to initiate, support, and monitor the fusion reaction. Heat produced by these systems, as well as the heat produced by the fusion reaction itself is collected by the ITER Cooling Water System (CWS) and rejected to the atmosphere. The CWS is composed of several systems designed for specific cooling roles. One of these systems is the Component Cooling Water System 2 (CCWS-2) whose function is to collect the heat from auxiliary client systems and components and transfer it to the Heat Rejection System. Clients have different requirements in terms of pressure, temperature, temperature variation, flow, and water quality. In addition they are located in different buildings located throughout the site and at varying elevations. To satisfy these different requirements the CCWS-2 is divided into four separate loops: CCWS-2A, CCWS-2B, CCWS-2C, and CCWS-2D. These loops vary in heat transfer capacity from 5 to 90 MW and vary in flow from 150 to 2000 kg/s. Each loop has different operating parameters. For example, the CCWS-2A loop is designed to cool components with wetted surfaces of copper and primarily serves the radio-frequency heating systems, magnet power supplies, and neutral beam injector system components. But even within a single loop, clients have varying requirements which challenge the system designers. This paper describes the evolution of the CCWS-2 system to match the needs of groups of compatible clients, and describes the development of the preliminary design one of its loops, CCWS-2A, to meet individual client needs.

P2-103 **Nuclear Analysis and Shielding Optimisation in Support of the ITER In-Vessel Viewing System Design**

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The In-Vessel Viewing System (IVVS) in ITER consists of six identical units which are deployed between pulses or during shutdown, to perform visual examination and metrology of plasma facing components. To protect the sensitive components of the IVVS from damage during operations, the system is housed in a shielded port between the vacuum vessel and the bioshield. CCFE has conducted a series of analyses with the aim of quantifying neutron flux and damage rates to the IVVS components and assessing the adequacy of several port shielding designs. Quantities such as damage and heating were required both during operation and at shutdown (due to the activation gamma source from irradiated material). In addition, the biological dose rate field from the activated IVVS in isolation was required to estimate transfer cask shielding requirements. A CAD model of the latest IVVS design (at the time) was simplified using the SpaceClaim software and converted to MCNP using MCAM. To ensure sufficient sampling of the IVVS vacuum vessel penetration, calculations were split into a 'global' and a 'streaming' contribution. The streaming contribution was modelled using the surface source feature of MCNP, whilst the global contribution to tallies was determined using CCFE's Global Variance Reduction (GVR) method. This approach allowed the IVVS penetration to be sampled ~104 times more than would have been possible in a conventional transport calculation, which in turn provided an improvement to the statistical accuracy of the results. CCFE's MCR2S software, a high resolution implementation of the rigorous two-step method, was used to determine the activation gamma sources in the IVVS and nearby ITER components for different scenarios and decay times. These sources were in turn used to determine the required shutdown dose quantities via additional MCNP transport runs. Results are provided and discussed here for the absorbed dose rate, nuclear heating and material damage rates in the sensitive piezoelectric motors, as well as for shutdown dose rate maps around the IVVS. The process of engineering optimisation of the port plug is also presented and discussed.

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P2-104 **RF coupler tests of the prototype RFQ linac for the IFMIF/EVEDA project**

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In the design of prototype RFQ linac for the IFMIF/EVEDA Project, a 175 MHz RFQ, which has a longitudinal length of 9.78m, was proposed to accelerate deuteron beam up to 5MeV. The operation frequency of 175MHz was selected to accelerate a large current of 125mA in CW mode. The overall driving RF power of 1.28 MW by 8 RF input couplers has to be injected to the RFQ cavity.For each coupler, nominal RF power of 153kW for CW mode and maximum transmitted power of 200kW for full reflection to be withstood up to 100ƒÊsec are required, and also maximum reflected power of 20kW has to withstand during RFQ operation with no beam. For these requirements, an RF input coupler with water-cooling port of a ƒÉ/4-long, including an RF window based on a 6 1/8inch co-axial waveguide, was designed. Three water-cooling channels are built into the loop antenna, the inner-conductor of RF window and the support disk, and the outer-conductor surrounding the RF window. On the production prototype, the flow rate of more than 5liter/min at the pressure of 0.3MPa was performed, it is found that total heat load of 10kW level can be removed. For the RF transmitted power test, a 10-sec CW operation using standing wave on a high-Q load circuit and a 1hour-CW operation with a coupling cavity are planned at Tokai-JAEA and INFN-Legnaro, respectively. These test results will be presented in details.

P2-105 **The simulation of seismic analysis for ITER fourth PF (Poloidal Field Coil) feeder**

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The ITER feeder systems connect the ITER magnet systems located inside the main cryostat to the cryo-plant, power-supply and control system interfaces outside the cryostat. The main purpose of the feeders is to convey the cryogenic supply and electrical power to the coils as well as house the instrumentation wiring. The Feeder carries superconducting busbars, supercritical cryo-pipes and instrumental pipes from the Coil-Terminal-Box and S-Bend-Box (CTB&SBB) to the coil. The PF busbar which carries 52kA current will suffer from high Lorentz force due to the background magnetic field inspired by the coils and the self-field between every pair of busbars. But to minimize the heat load to the busbars as well as the cryo-pipes, so the structure design proposed a balance between mechanical strength and thermal insulation performance have been achieved. Beyond that the dynamic mechanism on PF structure should be assessed. This paper presents the simulation and seismic analysis on ITER 4th PF feeder including the CTB&SBB, the Cryostat Feed-through (CFT), the In-Cryostat-Feeder (ICF), especially for supports of busbars and main cooling-pipes firstly. This analysis aims to study seismic resistance on system design under local seismograms with seismic amplitude, spectrum, duration and ground conditions comprehensively, the structural response curves and response duration results of displacement, tensile stress, and reaction force on supports under different directions actuating signals were obtained by using the seismic spectrum analysis and time-history analysis respectively. Based on the simulative and analytical results, the system seismic resistance and the integrity of the support structure in the 4th PF feeder have been studied and the detail design optimized.

P2-106 **Neutronic Analyses for the ITER electron cyclotron-heating upper launcher**

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The electron cyclotron resonance heating upper launcher (ECHUL) is installed in the upper port of the ITER tokamak thermonuclear fusion reactor for plasma mode stabilization (neoclassical tearing modes and the sawtooth instability). The paper reports the latest neutronic modelling and analyses which have been performed for the ITER reference front steering launcher design. The aim of the paper is to discuss results obtained for important nuclear responses such as neutron flux, heating and helium production which are shown to satisfy the nuclear criteria guidelines specified for ITER. Special focus is on the port accessibility after the reactor shut-down for which dose rate (SDDR) distributions on a fine regular mesh grid were calculated. The results are compared to those obtained for the ITER Diagnostics Upper Port. This analysis was performed on the basis of shut-down dose rate calculations employing the Rigorous 2-Step (R2Smesh) methodology which includes Monte Carlo MCNP5 transport calculations using FENDL-2.1 nuclear data and activation calculations with the FISPACT inventory code using EAF-2007 activation data. Use was made of accurate 3D models derived from original ITER and ECHUL CAD data through a dedicated interface software. High resolution SDDR calculations with the R2Smesh interface and the complex 3D B-lite neutronics model of ITER were possible by utilizing the HELIOS high performance computer at Rokkasho Japan. The calculations showed that the shut down dose rates satisfy the safety criteria specified for ITER, although the heterogeneous ECHUL design gives rise to enhanced radiation streaming as compared to the homogenous diagnostic upper port. This work is supported by Fusion for Energy under the contract No. F4E 2010 GRT 161. The views and opinions expressed herein reflect only the author's views. Fusion for Energy is not liable for any use that may be made of the information contained therein.

P2-107 **Improved Algorithms and Advanced Features for the CAD to MC conversion tool McCad**

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McCad is a geometry conversion tool developed at KIT to enable the bi-directional conversion of CAD and Monte Carlo (MC) geometry models for neutronics calculations (CAD to MC) and visualization purposes (MC to CAD). CAD software commonly adopt the BREP (Boundary-Representation) to represent the geometry. In contrast, MC codes adopt the CSG (Component Solid Geometry) representation which uses combinations of simple elements to describe a solid (such as primitive solids and half-spaces of analytic surfaces). The efficiency and reliability of the conversion process thus depends on the implemented BREP to CSG conversion algorithm. This work is devoted to an advanced version of McCad based on the implementation of an improved conversion algorithm. The conversion process builds on the decomposition of solids and the filling of the void spaces in between. These are two separate and complicated procedures which affect the conversion efficiency. New algorithms and functions have been developed for these two processing steps and implemented in a new version of McCad. In the decomposition process, a complex CAD solid is split into a number of simple convex solids. The employed decomposition strategy has been significantly improved in the new McCad version resulting in a more efficient conversion process. The void filling algorithm was analysed, completely changed and newly programmed. The new algorithm involves much less Boolean operations and hence is more stable, powerful and time efficient. Furthermore, a new user friendly interface for the material assignment in the MC geometry model has been developed and added to McCad as a new function. The new algorithms and McCad features have been successfully tested and applied for the CAD to MC conversion works on DEMO models, performed in the frame of the EFDA PPPT (Power Plant Physics and Technology) programme, and on NBI port duct liners for ITER.

P2-108 **Neutronic analyses of the HCPB DEMO reactor using a consistent integral approach**

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Within the EFDA programme on Power Plant Physics and Technology (PPPT) a conceptual design of a European DEMO is developed. Blanket concepts based on the HCLL, HCPB and WCLL designs are considered for the tritium breeding and power generation. Accordingly, a generic DEMO model is developed for the neutonic analyses, which serves as common basis for the integration of HCLL, HCPB or WCLL breeder blankets. The geometry model is based on a CAD model adapted for neutronic calculations using the Monte Carlo (MC) code MCNP. This model is converted to a MC geometry model by making use of a CAD to MC conversion tool. Such an approach satisfies quality assurance requirements, since it guarantees consistency of the CAD and the analysis model.

The generation of the new DEMO geometry model is performed in two steps. First the generic MCNP model with empty blanket space is automatically generated from the DEMO CAD model using the McCad geometry conversion software developed by KIT. Prior to the conversion step the CAD geometry model is slightly modified to adapt it to the MC neutronic simulation requirements. The automated material assignment is done at this point making use of the related new McCad function. In the second step, the empty space of the blanket modules is filled with the blanket internal structures of the different types making use of the very efficient repeated structure feature of the MCNP code. This step has to be done manually. The resulting MCNP geometry is compact, convenient and efficient for the neutronic analysis. In this work, the newly generated generic geometry model is used for parametric studies of the Helium Cooled Pebble Bed (HCPB) DEMO reactor. Three-dimensional Monte Carlo particle transport simulations are performed to this end with the MCNP5 code employing the full scale HCPB DEMO torus sector model with reflecting boundary surfaces.

Tritium breeding ratio calculations are performed to assess the tritium breeding performance, to determine the radial blanket build and the lithium enrichment which is required to achieve tritium self-sufficiency within a specified uncertainty margin. Various nuclear responses such as a neutron wall loading distribution, the power and neutron flux density as well as the nuclear power generation for the complete HCPB DEMO reactor are also provided.

P2-109 **Shut-Down Dose Rate Analysis for ITER Diagnostic Equatorial and Upper Ports**

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The purpose of this research consists in providing neutronics service for ITER Organization in developing of radiation shielding for the plugs inserted inside the equatorial and upper ports in ITER. One of important shielding criteria of the port plugs is human accessibility to the port interspace areas at the port back-sides, where the decision about the access granting is dependent on the value of Shut-Down Dose Rate (SDDR). The ability of the port plug designs to satisfy the radiation design requirements was analyzed in this paper, and design solutions for improving the shielding performance were recommended based on detailed neutronics calculations. Neutron and photon fluxes, nuclear heating, neutron damage, helium production were calculated along the ITER operation. In order to assess the port accessibility after the reactor shut-down, activation calculations and transport of decay photons were augmented to generate the SDDR distributions on high resolution meshes following to the Rigorous 2-Step (R2Smesh) methodology. This methodology includes radiation transport with Monte Carlo MCNP5 code using accurate 3D models derived from original ITER CAD data. Its activation calculations were performed with the FISPACT inventory code. The SDDR maps were visualized with the ParaView tool. To make the SDDR calculations with the R2Smesh interface possible for the complicated 3D B-lite neutronics model of ITER, the HELIOS high performance computational resources were engaged.

It was found that in the homogenized approximation of the port plugs, the dominant contribution to the SDDR inside the port interspaces is coming from the radiation streaming through the gaps surrounding the plugs, especially for the equatorial port. Therefore, the design shielding solutions have included a means to mitigate the gap streaming. To this end, the labyrinths of several configurations and horizontal rails inside the gaps were proposed. In estimations of shut-down dose rate in this paper only the contributions from the ITER activated materials were taken into account. The dose from airborne tritium and tritium contamination was not considered.

This work was supported by the ITER service contract Nr. 4300000548 for ‘Neutronics Analysis of ITER Diagnostics’.

P2-110 **An Exploratory Study on the Engineered Safety Features of a Fusion DEMO Plant**

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There have been numerous studies relevant to the safety of the fusion power generation with the tokamak concept including, but not limited to, those on the tritium behaviors, loss of in-vessel coolant accident (in-vessel LOCA), dust explosion, and accident scenarios and radiological consequences. The construction permit of the International Thermo-nuclear Experimental Reactor was issued by the French government after the review of the Preliminary Safety Analysis Report. The front-end studies on the safety issues of a fusion DEMO plant have been performed by the Korean DEMO team to define the design concept of a fusion DEMO, and discover the pathways to the fusion power generation with the tokamak concept. Even if a fusion DEMO reactor is postulated to be inherently safe, the need for some safety features to mitigate consequences of the incidents to minimize the occupational and residential radiation exposures is discovered with the review of the aforementioned safety studies. As the residual heats generated after the shut-down will not be so significant as to need an engineered system for the purpose of removing the decay heat, the elapsed time to reach the safe shutdown condition of the fusion DEMO reactor will be a governing factor to design the engineered safety systems. Mass-energy release rates as a function of the elapsed time to reach the safe shutdown are analyzed. Different sets of the engineered safety features which will ensure no need for an evacuation of the general public under the most severe in-vessel LOCA are proposed in consideration of the mass-energy release rates. A sensitivity analysis on the in-vessel physical conditions, where the fusion DEMO reactor will be assumed to be in the safe shutdown, is performed to review the inherent safety of the fusion DEMO reactor after a postulated in-vessel LOCA with the proposed engineered safety systems.

P2-112 **Effect of the Impurity on the Activation Analysis for the Korean HCCR TBM**

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Korea has been developing the Korean Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM) for an adoption in ITER. Based on the preceding neutronics and thermo-hydraulic analysis, a candidate design model has been developed. Using this design model, activation characteristics of the HCCR TBM was evaluated. The HCCR TBM uses mainly four TBM materials in each TBM component. The Reduced Activation Ferrite/Martensite (RAFM) steel is under development in Korea to be used as a structural material of First Wall (FW), Side Wall (SW), Back Manifold (BM) and Breeding Zone (BZ) plates. Li4SiO4 with the 6Li enrichment of 40% is being considered as a tritium breeder. Beryllium containing beryllium oxide (BeO) and graphite in SiC coating are used as a neutron multiplier and reflector material respectively. TBM materials contain impurities during a manufacturing process. Activation analysis considering impurities was performed in this study. The characteristics of the radioactive nuclides produced parasitically from impurities were evaluated. Then, the differences between the case with impurity and without impurity were compared in the view of the activities, decay heat and shutdown dose rate after a shutdown. Finally, effect of the nuclides which are dominant in the view of activation was estimated for the contact dose rate according to their contents. As a result, the effect of the impurity contained in the HCCR TBM materials was not significant in the total activip3-155ty. Only in the multiplier, the decay heat considering impurity was much higher than those without impurity until few days after a shutdown. The contact dose rates of the TBM according to the cooling time show that radioactive nuclides produced from normal composition are dominant up to one year, but dose rate due to impurity is significant after one year, especially for the RAFM steel.

Topic G Safety Issues

P2-113 **MFM-Based Diagnostic Technology for ITER DFLL-TBM System**

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Nuclear industries have increasing interest in using fault detection and diagnosis (FDD) methods to improve safety and availability of nuclear facilities. As a functional model, MFM (Multilevel Flow Models) is usually more explicit and simpler than other models used in real-time fault diagnosis. Based on the analysis of structures and functions of Chinese Dual Functional Lithium Lead Test Blanket Module (DFLL-TBM), a MFM model was established in this paper. With this abstract model, the diagnoistic process was been performed in three level: measurement validation (definded a set of measured flow values and used their redundancy to check consistency), alarm analysis (provided a set of alarm states which were associated with different parts of the MFM model, and the method can recognize primary alarms, while the other alarms were either primary or consequences of the primary ones.) and fault diagnosis (used the model to produce a backward chaining for diagnostic reasoning). These three diagnostic methods were validated in real cases. The results showed that MFM, with the graphic modeling and simple diagnostic logic, had obvious advantages in the fault diagnosis for nulcear equipment, and made sense for the development of a diagnostic expert system.

P2-114 **Evaluation of the neutron activation of JET in-vessel components following D-T irradiation**

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A novel Deuterium-Tritium experimental campaign is in preparation at the JET fusion facility for 2015. During the experiment, in-vessel components will be exposed to an intense high-energy neutron flux and neutron transmutation will thus generate a non-negligible activation of the machine components. The induced radioactivity and the decay after shutdown of the components depend on the material composition (including impurities), position and irradiation conditions. The JET components still today exhibit residual activation due to the passed experiments with deuterium-tritium fuel. The scope of the present work is to evaluate the time evolution of the activation of the main in-vessel components at the end of the future DT experiment. To carry out the activation evaluation it is necessary to calculate the neutron flux spectra in the selected components. This calculation is performed by the Monte Carlo MCNP5 code, which computes the neutron flux in the components of interest following the neutron emission in a typical DT plasma discharge using a three-dimensional model of the JET machine. The neutron spectra will be used as input for the FISPACT activation code. Different possible irradiation scenarios are considered in the investigation. In particular, in addition to the reference scenario, a shorter and a slightly modified campaigns are studied The results of the contact dose rates and specific activities for important nuclides presented in the paper will constitute the basis for the definition of the management strategy of the main activated in-vessel components following the D-T experiment. Furthermore the presented activity is preparatory to more detailed analyses that will provide comparison with the actual experimental data after the DT campaign.

* See the Appendix of F. Romanelli et al., Proceedings of the 23rd IAEA Fusion Energy Conference 2010, Daejeon, Korea

P2-115 **Tritium and Heat Management in ITER Test Blanket Systems Port Cell for Maintenance Operations**

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Six Test Blanket Modules (TBMs) will be tested in three equatorial ports of ITER, two TBMs per each port. Each TBM is part of a Test Blanket System (TBS) that includes also several ancillary systems located in the port cell and in other room of the Tokamak Complex. Each TBM port plug will be replaced several times during the ITER lifetime using a remote handling transfer cask during Long Term Maintenance (LTM) shutdowns, after the removal of the TBS components present in the port cell. Therefore, during each LTM shutdown, several operations have to be performed in the port cell (e.g., pipes cutting and re-welding) and human access is needed.

During operation, because of the high temperature of the TBS coolants, significant heat release occurs in the TBM port cells. Moreover, Tritium will also be present because of permeation and leakage from the tritiated fluids. It is required to keep the air temperature below 35°C, the concrete temperature below 50°C and to keep the Tritium-concentration below the admissible limit for human access for maintenance.

This study describes the computational fluid dynamics analyses that have been performed for the TBM port cells area for identifying the requirements in terms of heat power and Tritium extraction and for optimizing the design choices for achieving the objectives. The analyses have been performed for the port #16 because it features the highest level of Tritium permeation. The main results show that two local air coolers need to be installed in the port cell to extract about 11 kW and to simultaneously operate the detritiation system in the port interspace with an intake flow rate of 60 m3/h during operation with an additional 40 m3/h after shutdown for about half a day before starting maintenance.

P2-116 **Little Tritium Extraction System Pipe Break Environmental Impact by Atmospheric Modelling of Elemental Tritium Gas and HTO Transport**

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In the case of a little Tritium-Extraction-System (TES) pipe break (with critical failure of a fuelling line), the tritium source term has not yet been determined in the frame of European Test Blanket Systems, as Design Basis Accident (DBA) but it is expected to be in the order of a few grams. In this critical scenario acute modeling of environmental tritium transport forms (HT and HTO) for the assessment of fusion facilities dosimetric impact appears as of major interest. This paper considers different term releases of tritium-forms to the atmosphere from ITER which has experienced a frequent failure of a fueling line, due the little TES pipe break affecting a Helium-Cooled-Lithium-Lead Test-Blanket-Module. In case of 24.3 g of tritium were released from the broken fuelling-line directly into the gallery found only 0.5 g was released to the environment, assuming a little rupture in the TES piping located in the Port Cell. In this paper we assume a hypothetical daily release of one gram of tritium in HT and HTO forms. The daily failure is taken just in order to evaluate different meteorological scenarios or weather conditions.

The working model simulates the tritium forms dispersion and environmental impact out of the complex ITER-tokamak (and its safeguards) of selected environmental patterns both inland and in-sea using ECMWF/FLEXPART model. We explore specific values of this ratio at different levels. We examine the influence of meteorological conditions of the tritium behavior during 24 hours after the release. For this purpose we have used a tool which consists on a coupled Lagrangian ECMWF/FLEXPART model which is useful to follow real-time releases of tritium at low levels of the boundary layer to provide an approximation of tritium cloud behavior ranging from 3 to 24 hours. We have assessed HTO/HT ratios in a representative set of cases.

P2-118 **Visualized Nuclear and Radiation Safety Simulation Program and Its Application to Fusion and Fission**

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Due to the risks involved, work scenario in nuclear and radiation environment is always designed based on experts and past experience, without considering faults in plant design, human wrong operation by unskillful handing, risks associated with unpredictable situation. So the suggested work scenarios are always not the optimal scenarios according to as low as reasonably achievable (ALARA) principle. Virtual reality (VR) technology, in turn, has application in many diverse areas, with the possibility of performing virtual simulations of real environments. Then, it is possible to simulate these risk situations considering hypothetical scenarios. Based on VR technology and Rad-HUMAN, which is a whole-body voxel phantom of Chinese adult female developed by FDS Team, a visualized nuclear and radiation safety simulation program named RVIS has been developed for dose assessment and ALARA evaluation of work scenarios in nuclear and radiation environment. The basic functions of RVIS include: (1) CAD-based modeling and virtual assembly simulation of complex components; (2) real time simulation of multi-physical radiation process; (3) virtual roaming simulation and organic dose evaluation in radiation environment; (4) visualized analysis of dynamical 3D radiation field coupled with geometry model. The system architecture is based on Client/Servers model and it is composed of a parallel calculation server for large data calculation, such as real-time radiation simulation and 3D data rendering, and PCs for the three-dimensional visualization interface. RVIS allows the accurate assessment of organic dose rate by using Rad-HUMAN and the auto-optimization of worker scenarios based on multi-objective optimization algorithms. The system makes it possible to safely perform work scenario designs, optimization and pre-training of workers in risky areas. In this paper, the system architecture, ALARA evaluation strategy and some advanced visualization methods used in RVIS are described and an ITER maintenance analysis is shown for optimization of maintenance scenarios.

P2-119 **Safety managements of the IFMIF/EVEDA accelerator building**

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The Linear IFMIF/EVEDA Prototype Accelerator (LIPAc), the engineering validation up to 9MeV by employing the deuteron beam of 125mA are planning at the BA site in Rokkasho, Aomori, Japan. Since the LIPac is 1 MW huge beam-power accelerator, it is important to realize that the safety managements of IFMIF/EVEDA accelerator building to be considered the reduction of radiation exposure for personnel and of the radio-activation for accelerator components. For the safety managements, Personnel Protection System (PPS) and Machine Protection System (MPS) of LIPAc control system which work together with the radiation monitoring system and the access control system and etc. are applied. The safety managements are realized by three functions: access control, radiation monitoring and operation control of the LIPAc. The access control is mainly realized by PPS and eliminates the personnel exposure by accelerator operation. The radiation monitoring supervises the radiation leak, and if it is over the limit value, PPS will stop the LIPAc, immediately. In addition, the LIPAc control system receives the data of neutron and gamma-ray detector and provides these data to operators for supervision. Finally, the operation control supervises the LIPAc operation status, for example: operation time (total beam time), status of ventilation and etc. And, MPS inhibits the beam immediately if operation status became false. In this article, three functions to realize the safety managements will be presented in details.

P2-120 **Sensitivity study for a TES pipe rupture accident inside port cell**

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Korean Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM) for ITER is composed of four sub-modules and each sub-module has seven layers breeding zone, including three beryllium pebble neutron multiplier layers, three lithium ceramic pebble layers and graphite pebble layer as neutron reflector to avoid the use of a beryllium neutron multiplier. It is conceivable, however, when purge pipe guillotine break happens inside port cell (PC), graphite oxidation could take place in a TBM. Chemical reaction between air and breeding zone (BZ) materials can affect integrity of a TBM structure and it can lead BZ box failure and spitting pebbles to vacuum vessel. In this type of accident governing parameters are various, for example operating pressure, system volume, porosity of a pebble bed and material properties, etc. Thus, sensitivity study is an essential ingredient when attempting to determine proper design specification for Korean TBS. In this paper, based on preliminary accident analysis results for the current HCCR TBS, parametric study was performed. For this transient simulation, Korean nuclear fusion reactor safety analysis code (GAMMA-FR) was used.

P2-121 **Study on Safety Requirements of Korean Fusion DEMO Plant using Integrated Safety Assessment Methodology**

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In Korea, regulatory framework and safety requirements related with nuclear facilities is well organize through the operating and design experience of Light Water Reactor (LWR) since 30 years. So, same facility has been used to design and improve LWR. However, existing regulatory framework is focused on just LWR. In case of development of new reactors or plants, it may be inadequate to apply existing safety requirements, due to difference in coolant, fuel, and inherent safety feature of LWR. Development of Generation IV fission power plant (Gen IV) in Korea is one example. Currently, safety requirements which are adequate to Gen IV have been developed by using Integrated Safety Assessment Methodology (ISAM) as a tool. The ISAM consists of five analytical tools. These are Qualitative Safety features Review (QSR), Phenomena Identification Ranking Table (PIRT), Objective Provision Tree (OPT), Probabilistic Safety Assessment (PSA), and Deterministic and Phenomenological Analysis (DPA). Each tool is intended to address specific kinds of safety-related issues at different design phase. The resultant of each analysis tool support or relate to inputs or outputs of other tools. The ISAM is a PSA-based safety assessment methodology for Gen IV. Use of ISAM to establish safety requirements for Korean Fusion DEMO Plant (K-DEMO) may be possible because it is able to analyze the technical issues of varying complexity. The purpose of this study is to develop safety requirements of K-DEMO by using ISAM as a part of the R&D program funded by National Fusion Research Institute of Korea (NFRI). In the first part of this study, we have defined safety goals and principles and summarized each part of ISAM. Practically, it may not be possible to perform the analysis with all five tools because the development of K-DEMO is at preliminary conceptual design phase. So, the establishment of a part of safety requirements for K-DEMO adequately by using the results of QSR/PRIT or QSR/PRIT/OPT tools is discussed in the last part of the paper.

P2-122 **ITER Safety Studies: The effect of two simultaneous perturbations during a Loss of Plasma Control Transient**

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The loss of plasma control events in ITER are safety cases investigated to give an upper bound of the worse effects foreseeable from a total failure of the plasma control function. Conservative analyses based on simple 0D models for plasma balance equations and 1D models for wall heat transfer showed in the past that a hypothetical scenario of first wall coolant tubes melting and subsequent entering of water in the vacuum vessel could not be totally excluded. In a recent work[1], the synergistic effect of simultaneous overfuelling and overheating perturbations over ITER 500 MW inductive reference scenario during a loss of plasma control transient was shown, together with a way to assess the characteristics of the transients over a nT diagram. This contribution continues that work and presents new results on this topic, now exploring different combinations of perturbations and detecting the critical transients attending to the severity of their effects. The analysis of plasma-wall transients in this work is based in results from AINA code simulations. AINA (Analyses of IN vessel Accidents) code is a safety code developed at Fusion Energy Engineering Laboratory (FEEL) in Barcelona[2]. It uses a 0D-1D architecture, similar to that used for previous analyses of ITER loss of plasma control events. The results presented constitute a contribution to a possible improvement in the methodology used in the past for the ITER Safety study Loss of Plasma Control Transients.

[1] J.C. Rivas, J. Dies, "Safety studies: Review of loss of plasma control transients in ITER with AINA 3.0 Code", 27th Symposium on Fusion Technology, Liege, Belgium, September 2012

[2] J. Dies, et al., "AINA safety code, v1.0", Research Report, July 2007, 112p.

P2-123 **Neutron shielding and activation of the MASTU device and surrounds**

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MAST is a Spherical Tokamak that has been operating at Culham since 1998. A project to upgrade MAST is underway with the modifications being performed in a shutdown planned from October 2013 to March 2015. The neutral beam heating power will initially be ~5MW delivered from the existing 2 injectors and will be progressively upgraded to 10MW by the addition of 2 more injectors. Following the upgrade the duration of the plasma will progressively be increased from the current maximum of ~1s to ~5s. Consequently the neutron yield from each tokamak pulse, which is predominantly from heating beam ion interactions with the target plasma, is anticipated to increase by at least an order of magnitude.

For the greater neutron yields generated by these injected powers on-load neutron and photon doses were calculated for the previously existing shielding configuration using the radiation transport code MCNP. It was found to be necessary for the shielding to be increased to remain within agreed dose limits. Improvements to the shielding were recommended in various areas, including the following: shielding doors to MAST blockhouse; north access tunnel; blockhouse roof; west cabling duct.

Shutdown dose rates have been produced for both initial and ultimate stages of the MAST-Upgrade programme using the MCR2S code, which incorporates a rigorous two-step method for their calculation. These lead to recommendations on access control and on the planned maintenance regime. This work was funded [partly] by the RCUK Energy Programme under grant EP/I501045 and the European Communities under the contract of Association between EURATOM and CCFE. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Topic H FNT Special Neutron Sources

P2-125 **Characterization of MHD mixed-convection flows in a vertical rectangular duct with volumetric heating**

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We consider MHD liquid metal mixed-convection flows in vertical rectangular ducts with non-conducting walls in the presence of constant transverse magnetic field and volumetric heating. Such a flow occurs in the poloidal ducts of the DCLL (Dual coolant lead lithium) blanket with insulating flow channel insert, where lead-lithium (PbLi) is used as breeder/coolant and Ferritic steel as structural material. In these blanket flows, the forced flow is combined with the buoyant flow resulting in mixed-flow regime. Buoyancy effects in the liquid metal flow are caused by a non-uniform volumetric heating, whose intensity drops exponentially in the radial direction. The effect of buoyancy forces on the blanket operation manifests itself through additional thermal mixing, which can exhibit either laminar or turbulent features. The changes in the effective heat transfer coefficient/Nusselt number due to buoyancy effects may affect heat losses from the PbLi flow into the cooling helium streams. Also, knowledge of flow behavior is essential to further understand various phenomena such as tritium transport and corrosion of the Ferritic walls. In the present study, we carry out both analytical (1D) and numerical (2D and 3D) studies to better understand the flow behavior over a range of parameters: Hartmann (Ha ~ 40-1000), Grashof (Gr ~ 1e+06-1e+08), and Reynolds numbers (Re ~ 2000-10000). Among the most interesting findings are the tendency of the flow to two dimensionalization with increasing Ha, existence of two turbulence regimes, strong turbulence (ST) and weak turbulence (WT), and considerable variations in the Nusselt number. For undisturbed, fully developed flows there is a fair agreement between the 1D, 2D and 3D studies. Even though the parameter values considered in the present study are smaller compared to the blanket conditions the same tendencies of the flow behavior can be expected.

P2-126 **Experimental Validation of the ITER Blanket Attachments**

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Two main components of the ITER blanket attachment system are the flexible supports and key pads. This paper will present results of the mechanical testing of flexible supports and cylindrical key pads with a special attention to behavior of plasma sprayed ceramic insulation coatings and non-preloaded threads. The testing was performed in NIKIET in 2012-2013 during the final design stage of the ITER Blanket System.

The greatest design uncertainties of the ITER blanket attachments are associated with reliability and durability levels of ceramic electro-insulating coatings subjected to cyclic normal and tangent loads with cyclic sliding under unusually high pressure. As for today there is no approved structural design criteria and sufficient experimental data for ceramic electrical insulation under ITER-relevant EM and structural conditions. A goal of this experiment is to prove reliability and durability of ceramic coatings at flexible supports and key pads at ITER relevant loading conditions.

Another uncertainty is the fatigue lifetime of non-preloaded threads in the VV housing of the flexible support. There are no structural design criteria for the non-preloaded threaded joints working under the alternate axial load combined with cyclic bending moments neither in SDC-IC, nor in RCC-MR.

An unexpected result of key pad testing was an observation of a strong fretting effect, and, as a consequence, a generation of bronze dust. The fretting was associated with sticking between the pad and the key, associated with spikes at the loading diagrams and followed with loud bangs and strong vibrations. Further assessment is underway including the possibility of using a low friction coating to reduce this effect and will be reported in the paper.

P2-127 **Safety analyses in support of neutron detector calibration operations**

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5. See the Appendix of F Romanelli et al., Proc 24th IAEA Fusion Energy Conference 2012 San Diego, US

Neutron detectors in fusion devices need to be calibrated to provide the absolute neutron yield and the fusion power produced in fusion reactions. In large machines like JET, and even more in ITER, intense calibration neutron sources must be employed in order to produce statistically accurate signals in the detectors and in reasonable time intervals. In the new in-situ calibration of the JET neutron detectors in March-April 2013, a 252Cf neutron source is used with intensity of about 3×108 n/s. The source is introduced and deployed inside the vacuum vessel by remote handling. The source is delivered to the JET facility within a transport flask and the surface radiation levels must fall within transport regulations. However, some contingency scenarios may require transfer of the source into special shields: these are the operational shield and the auxiliary shield.

In this paper we describe the neutron calculations that have been carried out to evaluate the dose rate leakage from the shields which may contain the neutron source. The predictions are used to ensure safe working conditions for staff near the shields in particular circumstances. These include, for example, the transportation and receipt of source in the transport flask into the JET torus hall and the transfer of the source and its presentation to the robot which will extract the source and deploy it in the torus.

The calculations have been performed using accurate modelling of the neutron and gamma ray emission from the 252Cf source, and from the three shields. The differences on calculated dose rates deriving from the use of different flux-to-dose conversion factors have also been investigated. A comparison of dose rates calculated and measured is presented from the bare source (in cell) and with the source within its transport flask.

P2-128 **Development of Compact D-D Neutron Generator**

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Electrostatic accelerator based neutron generators are familiar for many factors like radiation safety, control operation, compact structure etc. One compact electrostatic accelerator based D-D neutron generator is under development in our laboratory. This generator has a hollow anode penning ion source that generates deuterium ions. These deuterium ions than accelerated and bombarded to a deuteriated titanium target. From the target due to the fusion reactions of deuterium particles, 2.45 MeV neutrons are generated. In this system the extraction and the acceleration of the deuterium ions is important. The ion source, which is used, is a self extracted ion source. After extraction, the ions are accelerated by one accelerating electrode. For steady state operation, it is necessary to stop the acceleration of the secondary electrons those are produced at the target due to bombardment of the deuterium ion, towards the ion source. Presence of these secondary electrons at the ion source may detoriate the performance of the ion source. For optimization of the ion optics, the transport of the ion beam is simulated by SIMION 8.0 software package. The ion extraction and acceleration as well as the suppression of the electrons at the target side have been taken care off. The generator has been designed and commissioned at our laboratory. The initial experiments have been carried out. The neutron yield was measured by bubble detectors. In the present experimental set up and operating parameters a neutron yield of 2 x 106 neutron/sec is achieved. In future experiments, we will endeavour for higher neutron yield. In this report, a detailed description about the generator, SIMION analysis and generation of the neutrons will be presented.

P2-129 **Parametrization of radiative properties of ICF mono- and multi-component plasmas**

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Plasma radiative properties, play a pivotal role in inertial confinement fusion. They are essential to analyze and explain experiments or observations and also in radiative-hydrodynamics simulations. Their computations require generation of large atomic databases and the calculation, by solving a set of rate equations, of a huge number of atomic level populations in wide ranges of plasma conditions. These facts make that, for example, radiative-hydrodynamics in-line simulations be almost infeasible. This has lead to develop analytical expressions based on the parametrization of radiative properties. However, most of them are accurate only for coronal or local thermodynamic equilibrium (CE and LTE, respectively). In this work we present a code for the parametrization of plasma radiative properties, in terms of plasma density and temperature, such as the radiative power loss, the Planck and Rosseland mean opacities and the average ionization for mono- and multi-component plasmas and plasma conditions typically found in inertial fusion confinement. Thus, as an example, we present in this work a parametrization of Au and Nd and their mixture (which are usually employed in the hohlraums in indirect drive ignition) and W and Xe and their mixture (elements that could be used in indirect drive ignition scheme as a constituent of the first wall of the reactor and as gas that protects the wall, respectively).

The databases of radiative properties for the parametrization are generated using the computational package ABAKO/RAPCAL [1]. Since the level populations are calculated using a collisional-radiative model, the calculations are valid for CE, LTE and non-LTE regimes. Furthermore, in order to ease the use of the parametrization code presented here this has been integrated on a user interface.

1.R. Rodríguez, R. Florido, J.M. Gil, J. G. Rubiano, D. Suarez, P. Martel, E. Mínguez, R.C. Mancini., "Collisional-radiative calculations of optically thin and thick plasmas using the computational package ABAKO/RAPCAL," Communications in Computational Physics, 8, 185 (2010).

P2-130 **Benchmarking of SuperMC2.0 with Fusion-Driven Subcritical System**

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Monte Carlo (MC) method is one of the most accurate computational methods in particle transport simulation. Some state-of-the-art codes, such as MCNP, GEANT4, FLUKA, EGS, TRIPOLI, are applied widely in simulating the interaction of particles with matter. Super Monte Carlo Simulation Program (SuperMC) developed by FDS Team is a multi-functional simulation program mainly based on Monte Carlo method and advanced computer technology. Firstly, constructive solid geometry (CSG) method was mainly employed to support geometry processes. Secondly, complete physics processes of neutron and photon of broad energy scale were considered, which were supported by Hybrid Evaluated Nuclear Data Libraries (HENDL). The SuperMC2.0 can perform neutron, photon and coupled neutron-photon transport simulation, with the criticality calculation and the tallies of cell flux, surface flux, surface current and energy deposition. The FDS-I, designed by FDS Team, is a Fusion-Driven Subcritical system, which consists of the fusion neutron driver with relatively easy-achieved plasma parameters and the subcritical blanket used to transmute long-lived nuclear wastes and to generate energy on the basis of self-sustaining of tritium needed for fusion reaction in plasma core and plutonium needed for neutron multiplication in the subcritical blanket. The major objective of FDS-I is to demonstrate the feasibility of early application of fusion energy technology. The plasma physics and engineering parameters of FDS-I are selected on the basis of considering the progress in recent experiments and associated theoretical studies of magnetic confinement fusion plasma and the progress in studies of blanket concepts optimization to reduce the requirement for neutron source intensity and subsequently plasma technologies. The neutronics model of FDS-I include nearly two thousand geometry cells. With the characteristics of both fusion and fission, the range of FDS-I¡¯s energy spectrum is larger than any other nuclear energy system. To verify SuperMC, a series of testing of international benchmark cases were used, and here showed a benchmarking to the FDS-I. The correctness of SuperMC2.0 was well proved by contrasting the results of other Monte Carlo codes. SuperMC2.0 can be well used for the design and analysis of FDS-I.

P2-131 **Support of Repeated Structure For Automatic Neutronics Modeling**

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The Monte Carlo (MC) method is widely used in nuclear analysis of complex nuclear devices. But modeling of complex geometries for MC computation is time-consuming and error-prone. Furthermore, the amount of cells may be beyond the processing capability of MC codes. Fortunately, many complex nuclear devices have repeated structures in their geometry. For an example, in International Thermo-nuclear Experimental Reactor (ITER) neutronics model, there are many divertor cassettes of the same shape. Repeated structure mechanism has been provided by many MC codes to support large-scale problems, which consist of many cells of the same shape.

This study proposed a method to convert Computer-aided design (CAD) models with repeated structure geometry to neutronics models in repeated structure format. This method has been implemented and integrated in the CAD/ Image-based Automatic Modeling Program for Neutronics and Radiation Transport (MCAM). And it has been validated by many testing examples such as ITER neutronics model. The result proved its correctness and effectiveness.

P2-133 **Benchmark Experiment on Titanium with DT Neutron at JAEA/FNS**

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Titanium is an important material for fusion reactor designs because it is contained in lithium titanate, which is a tritium breeding material candidate and is adopted in JADA test blanket module for ITER. In the nuclear design, accurate nuclear data are needed. However, few benchmark experiments had been performed for titanium. Thus we performed a benchmark experiment with a titanium assembly and a DT neutron source at JAEA/FNS in order to validate nuclear data libraries of titanium. The assembly was a titanium slab of 455*455*405 mm3 covered with 51 or 101 mm thick Li2O blocks, which reduced background neutrons inside the titanium slab. The assembly was irradiated at a distance of about 20 cm from the DT neutron source. Dosimetry reaction rates of the 93Nb(n,2n)92mNb, 27Al(n,a)24Na, 115In(n,n')115mIn, 197Au(n,g)198Au, and 186W(n,g)187W reactions were measured by the foil activation method inside the assembly. This experiment was analyzed by using Monte Carlo neutron transport code MCNP5-1.40 with recent nuclear data libraries of ENDF/B-VII.1, FENDL-2.1, JEFF-3.1.2, JENDL-3.3, and JENDL-4.0. The experimental assembly and DT neutron source were modeled precisely in the MCNP calculations. The JENDL Dosimetry File 99 was used as the response functions for the dosimetry reactions. The calculation result with ENDF/B-VII.1 agreed with the measured one the best. On the other hand, the agreement between the calculation results with FENDL-2.1, JENDL-3.3 and JENDL-4.0 and the measured one is not so good, particularly for the reaction rates of the 197Au(n,g)198Au, and 186W(n,g)187W reactions which are sensitive to lower energy neutrons. The calculation result with JEFF-3.1.2 shows the similar tendency as that with ENDF/B-VII.1, but note that a part of nuclear data of titanium isotopes in JEFF-3.1.2 is substituted for that of natural titanium in ENDF/B-VI and is not evaluated for each isotope. The details will be presented in the symposium.

P2-134 **Effect of impurities on vacancy migration energy in Fe-based alloys**

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Reduced-activation ferritic/martensitic steels have been developed as the prime candidate materials for experimental fusion reactors. To estimate irradiation effects in fusion reactor components, multiple-scale modeling has been studied. Modeling activities for irradiation induced microstructural change is quite effective to enhance the capability to predict mechanical properties of the materials during irradiation. Defect activation energies such as vacancy and interstitial migration energies should be estimated to obtain fundamental parameters for the modeling. In this study, pure iron and Fe-based model alloys including carbon and nitrogen have been irradiated by electron and ion beams using a high voltage electron microscope in order to clarify the effect of carbon and nitrogen on microstructure evolution and defect activation energies. Growth rate and saturated number density of dislocation loops have been measured and Arrhenius plotted to estimate the migration energies of point defects. Electron irradiation experiment indicated that the net migration energy of vacancy in a high purity iron tended to be lower compared to that in low purity pure iron and model alloys. Furthermore, vacancy migration energies in all the specimens including more carbon was slightly higher than that including more nitrogen.

P2-135 **Conceptual design of a helium heater for high temperature applications**

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The Helium-cooled Modular divertor with Jet cooling (HEMJ, 10 MPa, 600/700 °C) has been developed in Karlsruhe Institute of Technology (KIT). In order to provide appropriate testing conditions for the divertor modules as well as qualify the materials for high heat flux, high temperature and high pressure applications, the Karlsruhe Advanced Technologies Helium Loop (KATHELO) has been designed. The loop, currently under construction, operates conditions at pressures as high as 10 MPa, and temperatures up to 800 °C. In order to achieve the required temperature level a 200 kW electrical heater is installed before the test section. In order to cope with the extreme values for the design conditions, the heater has been designed with two vessels: an external one that forms the pressure barrier, while the heater itself is installed in a second vessel installed inside the first one. The pressure vessel is maintained at low temperature by the helium coming from the circulator (~70 °C) and it can be designed like a conventional unit. The inner body will be operated at high temperature but it is no longer a pressure vessel (it acts more like an in-liner) and, therefore, the thickness of the mantel can be reduced significantly. In this paper the conceptual design of the unit will be presented and the impact of the coupling between the cold and hot helium gas on the overall efficiency of the loop will be investigated. In addition, a detailed thermal dynamic and stress analysis of the feed through of the hot helium (750 °C) into the low temperature pressure vessel will be presented using ANSYS. Finally, the heater will be implemented in the helium loop of KATHELO using RELAP5-3D that the system behavior of high temperature operation will be presented as well.

P2-136 **Study of the sensitivity of a Cerenkov Fibre Optics Sensor (C-FOS) in the IFMIF Test Cell**

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An optical-based detector, named C-FOS (Cerenkov Fibre Optics Sensor), is intended to be included in the Start Up Monitoring Module, STUMM, which will be used during the IFMIF (International Fusion Materials Irradiation Facility) commissioning phase. That sensor must support the harsh conditions in the IFMIF irradiation area in terms of radiation, temperature and strong space restrictions. The design of the detector is being developed in collaboration with SCK·CEN within the framework of the EVEDA (Engineering Validation Engineering Design Activities) phase of IFMIF. CIEMAT is in charge of the study of the detector sensitivity, including neutronics calculations.

Since the Cerenkov Effect is strongly directional and the IFMIF neutron source is quasi-directional (mostly parallel to the deuteron beams), numerical simulations are necessary in order to evaluate the feasibility of the diagnostic. A C-FOS model, consisting of an optical fibre made of silica, with a certain diameter, has been included in the McDeLicious code. McDelicious is an enhancement to the MCNP5 code in order to adapt computational requirements of the IFMIF Test Cell. The fibre occupies the entire footprint region, being its axis perpendicular to the IFMIF beam direction. The number and distribution of Compton electrons generated in the detector when the IFMIF accelerator operates at full power is calculated. These electrons are the responsible of the Cerenkov light which can be partially collected by the optical fibre. The angular dependence of Cerenkov photons in the C-FOS is also studied, being mainly sensitive to an efficiency parameter, η, which describes the coupling between the generated photons and the numerical aperture, NA, of the fibre. Different configurations of the fibre inside the irradiation modules are shown, as well as the dependence with different parameters which affect the C-FOS signal. This work has been funded by the Spanish Ministry of Economy (MINECO) under projects AIC10-A-000441 and AIC-A-2011-0654.

P2-137 **Design prospect of remountable high-temperature superconducting magnet**

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The remountable (mountable and demountable) high-temperature superconducting (HTS) magnet has been proposed for future fusion reactors, segments of which are mounted and demounted repeatedly. This attractive concept could be very helpful for improved reactor maintenance and/or construction of the large, complex superconducting magnets required for fusion reactors. For example, the concept has been proposed for application to a component-testing machine (small tokamak), Vulcan, and a heliotron-type fusion DEMO reactor, FFHR. We have carried out R&D activities on the remountable magnet; developments of mechanical joint methods of HTS conductors and a high heat removal technique using cryogenic coolants, structural and thermal designs of the remountable magnet. This paper summarizes recent R&D activities and shows prospect of the design. We have been tested three joint configurations; mechanical lap, butt and edge joints. We developed a method of evaluation of joint resistance depending on joint configurations, current, temperature, strain and magnetic field based on numerical and experimental evaluations. The joint configurations have been improved and optimized with the above evaluations. We also demonstrated the low joint resistance, of the order of 0.1 nano-ohms using 30 kA-class HTS conductor samples. The joint resistance can be acceptable from a viewpoint of electric power for cooling the magnet. In addition, we have tested heat removal technique using metal porous media and cryogenic coolants. The technique can achieve larger critical heat flux (CHF) and heat transfer coefficient with latent heat and large area of heat transfer surface. We demonstrated CHF of 0.755 MW/m2 using bronze sintered porous media and subcooled liquid nitrogen. We also conducted thermal analysis to obtain temperature field of the remountable magnet depending on cooling techniques, coolants�f temperature and the magnet configuration. Based on the above investigation, we propose some configurations of the remountable magnets having low joint resistance and high heat removal performance.

P2-138 **Molecular dynamics simulations to evaluate the nano-mechanism responsible for irradiation hardening in alpha-iron**

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Mechanical property changes are a key factor in determining the lifetime of fusion reactor materials. Irradiation hardening, especially in BCC alloys, is an important property change to consider for fusion materials. The 14MeV neutrons generated by D-T reactions create irradiation-induced defects, and their interaction with a line dislocation is one of the micro mechanisms behind irradiation hardening. In BCC alloys, some of the defect clusters are glissile in nature, moving to stable positions in the strain field of a line dislocation, a long-range interaction. Other clusters intersect directly with the moving dislocations, becoming obstacles to dislocation motion, which is a short-range interaction. In order to construct a comprehensive model describing irradiation hardening, we develop a method for evaluating both long and short range interactions of a line dislocation and irradiation-induced defect. Molecular dynamics (MD) is used as a tool to evaluate the short-range interaction, whilst the long range interaction is treated by elastic theory. In our previous work, results obtained from the elastic interaction are in good agreement with the MD results for the stable positions of a cluster near a line dislocation.

In this study based on previous work, MD simulations are conducted to evaluate the interaction between a line dislocation and a cluster and clarify the possible mechanisms of irradiation hardening in alpha-iron.

When the centroid of a cluster lies just on the slip plane of an edge dislocation, the cluster moves to and collides with the edge dislocation even without external stresses. Then, it splits into two smaller clusters. One on the extra half plane side quickly escapes, and the other remains completely absorbed. A screw dislocation tends to absorb a cluster by cross-slip and forms a helical structure along a line dislocation, which appears to be a strong obstacle for dislocation motion under an applied stress.

We will discuss the microstructural evolution processes and possible hardening mechanisms under irradiation in alpha-iron.

P2-139 **Verification of the displacement damage processes by STM observation at atomistic spatial resolution**

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A multi-scale modeling or simulation is normally used in order to predict the material behavior under the fusion environment with 14MeV neutron irradiation. This approach involves a displacement damage in high energetic particles by molecular dynamics simulations (MD), diffusion and interaction of irradiation-induced defects and/or dislocations by kinetic Monte Carlo simulations (kMC), defect cluster nucleation and growths kinetics by rate theory approaches, and change in the mechanical properties by molecular dynamics simulations, sometimes with dislocation dynamics simulations. However, there have not been sufficient experimental facilities to verify such kind of simulations, especially for MD and/or kMC simulations, due to the limitation of the spatial resolution, which sometimes may obstruct further understanding of the material behavior under the fusion environment.

We have constructed a linked facility of a scanning tunnel microscope (STM) and an ion accelerator to observe surface defect formation at atomistic spatial resolution. With this facility, we can observe the ion-irradiated samples by keeping it in an ultra high vacuum at about 10-7 Pa in order to avoid surface oxidation. We also can change the ion energy from 10 keV up to 50 keV, and can anneal the irradiated specimens at the same chamber. We prepared the samples of Au(111) with reconstructed surfaces and performed the Ar ion irradiation at liquid nitrogen temperature. After ion irradiation, we observed the surface to detect the number and morphologies of irradiation-induced defects at atomistic spatial resolution. We also conducted the annealing experiments for the irradiated samples to analyze the recovery behavior. Irradiation-induced defects cluster are clearly observed on the irradiated surfaces. The maximum size is much larger than that calculated by the conventional approaches. It is apparent that there is a certain mechanism for defect formation other than direct collision with high energetic particles.

We will compare the results for displacement damage processes by MD simulations, and discuss the validity of these simulations.

Topic I Repair and Maintenance

P2-140 **Development of a brazing connector for DEMO in-vessel components**

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The cost of electricity generated by fusion power will be strongly conditioned by the availability of future reactors. One key issue is the developing of feasible quick pipe connectors for the connection/disconnection of critical in-vessel components as breeding blanket or divertor during maintenance operations.

Brazing is a widely used joining technique which produces leak-proof high strength joints, with excellent stress distribution and little distortion. Specifically, induction brazing provides control of the heat pattern and fast and located heating, which avoids or minimizes oxidation and damage in previously heat-treated base materials.

This work presents a design of a self-brazing/debrazing connector to be used with helium, lead-lithium and water pipes in DEMO. The remote handling compatible design includes an induction heating system, a brazing atmosphere supply, an inspection system (leak testing), a bolted/clamped union to provide stiffness against disruptions and thermal loads, and a positioning and alignment system. Results from electromagnetic-thermal and capillary flow analyses for evaluating the performance of the connector are discussed. The issue of tritium permeation, as well as the phenomenon of transmutation in the connector materials for the expected DEMO irradiation scenario is also assessed.

P2-141 **Maintenance Duration Estimate for a Fusion Power Plant based on the EFDA DEMO 2012 design concept**

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The high neutron flux in a fusion power plant results in the need for frequent remote replacement of the plasma facing components. The rapid and efficient exchange of these components is therefore a key element to the success of a fusion power plant. The maintenance duration for the remote replacement of the shielding and tritium breeding blankets and the divertor cassettes has been estimated for a power plant based on the EFDA DEMO design concept for 2012, with data extrapolated from recorded times and operational experience from remote maintenance activities on the JET tokamak. The results suggest that for a highly developed and tested maintenance system with a large element of parallel working and challenging but feasible operation times the replacement of the blanket and divertor components could be achieved within the desired time frame of 6 months. This estimate is based on the required maintenance activities for the current pre-concept remote handling system design and the duration is bound to increase as the design develops and the complexity increases. Other aspects to be managed during the system development includes; provision of the resources required to develop the complex maintenance system to the required level, the ultimate reliability that can be achieved for the remote handling equipment and processes and meeting the requirements of safety legislation.

Key words: DEMO, remote handling, maintenance time, operational efficiency

This work was funded by the RCUK Energy programme and by EFDA

P2-142 **Concept for a Vertical Maintenance Remote Handling System for Multi Module Blanket Segments in DEMO**

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The anticipated high neutron flux and the consequent damage to in-vessel components in DEMO results in the need to regularly replace the tritium breeding and shielding blankets. A number of proposals to implement this exchange have been proposed, for example: large port maintenance systems—such as DREAM, CREST, SlimCS, and the US ARIES-RS and ARIES-AT—, and vertical maintenance systems—namely NET and other European concepts— where vertically arranged multi module blanket segments are extracted through a vertical access port.

There are many merits to the vertical maintenance system, including: smaller toroidal field coils, simplified coil structural supports, smaller vacuum vessel closure plates, and reduced size and scope of containment cask handling and hot cell activity. This system, however, does require complex in-vessel and ex-vessel remote handling equipment.

This paper presents novel conceptual designs for a set of remote handling equipment for the vertical maintenance scheme. This incorporates designs for an in-vessel transporter, to detach and attach the blanket segments, and cask-housed vertical maintenance devices to open and close access ports, cut and weld service connections, and extract blanket segments from the vessel. The ability for these items to withstand the in-vessel environment is discussed. Areas of further study are identified in order to comprehensively establish the feasibility and evaluate the efficiency of the proposed maintenance system.

Key Words: DEMO, remote handling, blanket, vertical maintenance system. This work was funded by the RCUK Energy Programme and by EFDA.

P2-143 **Progress in the design, R&D and procurement preparation of the ITER Divertor Remote Handling System**

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The ITER Divertor system is segmented into 54 parts, also known as Divertor Cassettes. The full replacement of the Divertor system is a scheduled maintenance task that will take place between 2 and 3 times during the lifetime of the ITER machine and therefore is key factor for the reactor availability. The ITER Divertor Remote Handling System (DRHS) consists of a number of dedicated remote handling equipment and tooling that will provide the means to perform the exchange of the Divertor system in a full-remote way. The novelty and complexity of the remote operations (such as millimetric positioning of the cassettes, cutting and re-welding of cooling pipes in a nuclear, dark and geometrically constrained environment), together with strict RAMI and safety requirements, constitute a major challenge of the ITER Remote Maintenance System. Fusion for Energy (F4E), i.e. the European ITER Domestic Agency, is responsible for organizing and carrying out the procurement of the ITER DRHS. The procurement of the DRHS is based on a functional set of system requirements and supported by a reference design (developed at a conceptual level). The reference design of the ITER DRHS has been the result of an extensive R&D programme (coordinated by IO, F4E and EFDA) and fulfilled by a number of European fusion laboratories and industries. This paper first presents a technical description of the ITER DRHS, including the main system requirements (functional, operational, safety, RAMI) and the resulting conceptual design (made of cassette movers, dexterous robotic arms, end-effectors, tooling, and control system). The second part of the paper introduces the procurement roadmap towards the delivery of the ITER DRHS to the IO by the year 2020 and to the more recent progresses of the project.

P2-144 **DEMO Active Maintenance Facility Progress 2012**

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In Europe the work on the specification and design of a Demonstration Power Plant (DEMO) is being carried out by EFDA in the Power Plant Physics and Technology (PPP&T) programme. DEMO will take fusion from experimental research into showing the potential for commercial power generation.

This paper describes the progress being made in the conceptual design of a DEMO Active Maintenance Facility (AMF). The AMF would used for the storage, handling and processing of In-Vessel Components (IVC) throughout their time on site with the only exception being the time they are installed in the vessel. The high levels of activation expected for the IVCs means all handling operations associated with used components will have to be carried out using remote handling techniques.

Along with the high level of activation, the IVCs also have high radiation dose rates and decay heating. Both of these give problems in the AMF. The high dose rates require the remote handling equipment to be sufficiently radiation tolerant to allow it to work reliably for long periods. The decay heating requires forced cooling of newly removed IVCs while they are in storage. The duration of the storage is dependent on the decay heating reducing to a level that is more compatible with dextrous remote handling. This level has been nominally set at the point where the IVC will be <50°C with out active cooling in room temperature air. The storage capacity, and the efficiency with which it can be accessed, would be a key feature of the AMF. For DEMO the storage duration requirements are simpler as there is sufficient time to process the used IVCs before the space is required for the next set to be exchanged. In a power plant there would be a trade off between processing capacity and storage space. It is assumed that the speed at which the AMF could receive and dispatch components would be sufficient to keep it off the critical path of a planned shutdown. This would have to be validated when accurate estimates of the removal and installation process times are made.

P2-145 **ITER divertor Thomson scattering system: in-vessel movements and remote handling kinematics**

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One of the main challenges for many ITER diagnostics is in-situ movements of diagnostic components wherever they might be used: shutters, adjustment or broken component replacement. The divertor Thomson scattering diagnostic design includes the laser mirror protected by a window, mechanisms for replacement and for shuttering. Laser launcher is one of the most challenging elements of DTS design. The front end laser optics must preserve optical properties under laser radiation, heat loads, neutron, gamma, X-ray, UV irradiations. This paper describes a specially developed remote handling actuator technology and different in-situ movable mechanisms for alignment, shuttering or replacement of broken optical elements. The specially developed actuator based on application of toroidal magnetic field has been developed, prototyped and tested. The developed linear actuator should be used in push/pull operation in challenging ITER conditions e.g. vacuum 10-6torr, eddy currents induced by plasma disruptions, neutron fluxes and heating up to 250^{o}C during baking procedures. Regular inspections and maintenance of ITER divertor requires use of remote means based on robotic technologies that enable extension of human capabilities into the machine. Savings on costs requires development of remote handling (RH) port diagnostic equipment intended for the divertor cassettes RH robot capabilities.

The installation/uninstallation procedure of the equipment located in the divertor RH port #8 must be adopted for the following RH robot capabilities: manipulation and exchange components weighing up to 50tons; moment of rotation of a special RH arm tool 20Nm, welding, including tubes with 72mm have been developed.

P2-146 **Design of the remote handling equipment for the refurbishment of the European target assembly design for IFMIF**

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The remote handling maintenance of components of the IFMIF facility is one of the most challenging activities to be performed to guarantee the required high level of IFMIF plant availability. Among these components the maintenance of target assembly system appears to be critical because it is located in the most severe region of neutron irradiation. The present European target assembly design is based on the so called replaceable backplate bayonet concept. It was developed with the objective to reduce the waste material and to simplify the procedures for the target and backplate replacement and thus reducing the intervention time for their substitution. The remote handling maintenance activity of the target assembly comprises a number of in situ refurbishment tasks, like the: removal of the backplate, cleaning of surfaces from lithium solid deposition, inspection of the target body, installation of a new backplate and testing of the assembled system. However there is also the possibility to replace the entire target assembly and to perform these refurbishment tasks online in a dedicated hot cell. To accomplish all the refurbishment operations for the target assembly within the expected time for maintenance, the annual preventive maintenance period for IFMIF has been fixed in 20 days, a set of remote handling equipment and tools has been developed and tested in ENEA Brasimone. The design of these equipment is described in the paper together with the procedures for the refurbishment of the target assembly.

P2-147 **Hardware availability calculations and results of the IFMIF accelerator facility**

Bargalló, Enric[1]; Arroyo, Jose Manuel[2]; Abal, Javier[1]; Beauvais, Pierre-Yves[3]; Gobin, Raphael[3]; Orsini, Fabienne[3]; Weber, Moisés[2]; Podadera, Iban[2]; Grespan, Francesco[4]; Fagotti, Enrico[4]; Dies, Javier[1]; Tapia, Carlos[1]; De Blas, Alfredo[1]; Mollá, Joaquín[2]; Ibarra, Ángel[2]

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Hardware availability calculations have been done individually for each system of the deuteron accelerators of the International Fusion Materials Irradiation Facility (IFMIF). The principal goal of these analyses is to estimate the availability of the systems, compare it with the high IFMIF availability requirements and find new paths to improve availability performances. The parallel activity on the design and construction of the Linear IFMIF Prototype Accelerator (LIPAc) provides detailed design information for the RAMI (Reliability, Availability, Maintainability and Inspectability) analyses and allows finding out the improvements that the final accelerator could have. Because of the R&D behavior of the LIPAc, RAMI improvements could be the major differences between the prototype and the IFMIF accelerator design. Major unavailability contributors are highlighted and possible design changes are proposed in order to achieve the hardware availability requirements established for each system. In this paper, such possible improvements are implemented in fault tree models and the availability results are compared. These analyses have been performed in collaboration between system designers and RAMI experts enabling the creation of RAMI models that reflect the current accelerator design and allows to propose plausible design modifications in order to improve the availability performances of the machine.

P2-148 **RAMI status in the IFMIF Test Facilities at the end of the engineering design phase**

Abal, Javier[1]; Dies, Javier[1]; Arroyo, Jose Manuel[2]; Bargalló, Enric[1]; García, Ángela[2]; Casal, Natalia[2]; Mas, Avelino[2]; Tapia, Carlos[1]; De Blas, Alfredo[1]; Mollá, Joaquín[2]; Ibarra, Ángel[2]

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The International Fusion Material Irradiation Facility, IFMIF, has the aim to provide a neutron intense source to test and qualify materials for the future fusion reactors. The 96% availability requirement allocated to the IFMIF's Test Facilities has been a critical constrain in the engineering design phase. The RAMI (Reliability, Availability, Maintainability and Inspectability) activities have been monitoring the design and proposing changes in order to achieve the IFMIF high availability requirements.

This paper shows how the availability in all Test Facilities' systems -especially in the Test Cell- has increased during the RAMI oriented design evolution. The Failure Mode, Effects Analyses, FMEAs, focused on downtime have been developed into fault and event trees. Their results have been used as a tool to improve specific maintenance strategies and management practices of keystone components. The final availability allocation implementation endorses the reliability requirement to each component and highlights special tests in future prototypes when needed. As a result, the RAMI activities have demonstrated a strong performance when being implemented in parallel to other engineering design activities.

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P2-149 **Availability simulation software adaptation to the IFMIF accelerator facility RAMI analysis**

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The commercial reliability software RiskSpectrum® PSA Professional has been used for the Reliability, Availability, Maintainability and Inspectability (RAMI) analysis performed for the International Fusion Material Irradiation Facility (IFMIF) accelerator. This software allows making powerful statistical analyses to obtain detailed and specific results in order to improve the design of the accelerator systems. However, as the model grew and the complexity was increased, some aspects became difficult to be modeled. When operation requirements and first maintenance policies appeared, a simulation of the whole performance of the accelerator became interesting. A simulation allows taking into consideration relevant parameters and complexities that reflect the behavior of the whole accelerator in a better way than commercial software packages can do.

Availability simulation software used for the International Linear Collider (ILC) became an excellent option to fulfill RAMI analysis needs. This software, called AvailSim, was used not only for designing ILC but also for other accelerators. Nevertheless, this software needed to be adapted and modified to simulate the IFMIF accelerator facility in a useful way for the RAMI analyses in the present design phase. Furthermore, some improvements and new features have been added to the software. This software has become a great tool to simulate the peculiarities of the IFMIF accelerator facility allowing further changes or improvements to obtain a realistic availability simulation. Degraded operation simulation, manpower management and maintenance strategies are the main relevant features. In this paper, main AvailSim software modifications and adaptations to IFMIF RAMI analysis are described. Moreover, first results obtained with AvailSim and a comparison with previous results are shown.

P2-150 **Improving the performance of DTP2 bilateral teleoperation control system with haptic augmentation**

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Remote maintenance can be a challenging and mentally demanding job for the operators of bilateral teleoperation systems. It is especially difficult in an environment such as the ITER divertor region where the maintenance tunnels are confined, pitch black and the operators have to be able to operate heavy loads and implement delicate tasks. Also, because of the radiation during maintenance, only a limited number of radiation tolerant cameras can be used for video feedback. Deployment of these cameras for optimal field of view is a tedious task, rendering the available video feedback poor.

Computer aided teleoperation (CAT) systems can reduce the amount of mental and physical work load perceived by the operators significantly and make teleoperation tasks faster, by haptically assisting the operator teams. An experimental CAT sub-system was developed at the DTP2 (Divertor Test Platform 2) to assist the divertor remote maintenance operator team in the divertor cassette locking sequence (CLS). The DTP2 CAT system offers virtual fixture based haptic assistance to the operator by guiding the remote manipulator to specific points of interest in teleoperation workspace and preventing unintended collisions. An experimental study was concluded to evaluate the achieved increase of operator performance with CAT in one of the divertor CLS steps. In the experiment 10 test operators repeated a CLS related tool pick-up task using a full size 6DOF water hydraulic manipulator, designed for the divertor maintenance. Each operator performed the task twice, once with the assistance from CAT system and once without it. Operator actions were recorded during the tasks and questionnaire method was used to evaluate perceived task load of the operator immediately after the tests. The study revealed a statistically significant performance increase in both the execution times of the teleoperation task and operator comfort levels during tasks when CAT aids were in use.

P2-151 **Design of Structural Components for the helical DEMO reactor FFHR-d1**

Tamura, Hitoshi; Goto, Takuya; Tanaka, Teruya; Masuzaki, Suguru; Yanagi, Nagato; Miyazawa, Junichi; Sagara, Akio

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FFHR-d1 is the conceptual design of the helical DEMO reactor being developed at the National Institute for Fusion Science. The maintenance of in-vessel components is very important for the fusion demo reactor. In addition, sufficient paths are needed for the diverter exhaust. To solve these problems, the vacuum vessel, a coil support structure, and a cryostat need large apertures. The helical-type reactor has attractive features that there is a wide space between the helical coils. This space can be useful for@ the maintenance. However, the coil support structure has to be sufficiently rigid to stay within soundness and deformation limits. The structural design of each component has to be decided considering the deformation and the stress/strain caused by environmental conditions such as temperature change, induced electromagnetic force, and support method. In addition, the weight of the coil support structure including the superconducting coil is approximately 20,000 tons and that of the blanket system is tens of thousands tons. How to support this weight while ensuring adiabatic conditions using a cryostat, what plays a role in not only maintaining the vacuum boundary but also being the supporting base for the interior equipment, are also critical. A design combining the structural components in FFHR-d1 was conducted from mechanical and thermal viewpoints. As the result, sufficiently large port area of 8.5 m x 7.5 m has been ensured. The maintenance and exhaust schemes have been investigated according to this fundamental design.

P2-152 **A R&D program on Leak localization concepts for actively cooled fusion machines**

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High vacuum conditions are required for the fusion machines operation. Any leak in the vessel, even micrometric (down to 10-6 Pa.m3/s) doesn't allow their operation. The conditions for the leak localization in the next step actively cooled machines like ITER will be more severe and human intervention will no longer be possible. Thus the techniques used in the current actively cooled tokamak like Tore Supra will not be relevant. These techniques consist mainly in pressurization/depressurization cycles of the sub circuits feeding the Plasma Facing Components (PFC). From the Tore Supra experience, some statements can be made. The localization process must be considered in early stage of the cooling circuit design in order to be able to mitigate the leaky PFC. Leak Localization diagnostics which could be quickly deployed as soon as a leak appears are needed. A R&D program on this last topic has been launched at CEA/IRFM since 2009. Its objective is to define the best experimental conditions and to select the most efficient concepts or diagnostics to localize helium or water leaks quickly maybe by remote operation. Their performances should allow detecting in a few seconds traces of helium or water close to the first wall, corresponding to a micro-leak. Leak sensors would have to be light weight devices in order to be integrated on a carrier or as small as possible to be integrated into the first wall, both of these options should be taken in consideration. The specifications for the Leak Localization concept will depend on the constraints of each machine. In a first step, the concepts must be selected among the available technologies and the specific conditions for their implementation. In this framework, a compliance matrix concept was used. In a second step, the concepts must be validated and qualified in relevant conditions. Two test-beds named STILL (Sensors Tests for ITER Leak Localization) and ROVE (Remote Operated Vacuum Equipment) have been erected and qualification tests were performed. The complementarity of the two test-facilities allows answering the most of identified issues for the leak localization. The paper will describe some of the most promising concepts, the selection methodology, the test-facilities and the results obtained so far.

P2-153 **RAMI analysis in IFMIF remote handling operations**

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The International Fusion Materials Irradiation Facility, IFMIF is considered in the international community as an essential milestone to achieve future nuclear fusion power plants. Its operational availability is one of the most important constrains in order to provide timely information on materials behavior under relevant irradiation conditions. The remote handling (RH) maintenance system is critical in achieving IFMIF's high availability requirement as failures in its components or in the maintenance interventions may develop into high downtimes.

In this paper, a RAMI (Reliability, Availability, Maintainability and Inspectability) analysis of the different remote handling operations is presented focused in reducing the downtime on the scheduled and unscheduled maintenance. The use of reliability techniques as FMECA analyses highlights human influence as an essential factor in these tasks. Therefore, the common RAMI approach has been merged with the HRA (Human Reliability Assessment) by the HRA THERP (Technique for Human Error Rate Prediction) method. The results show the major unavailability contributors in the RH operations and the directions to counteract them; hence this analysis is a successful tool to increase availability from a design stage. This work has been funded by the MINECO Ministry under projects AIC10-A-000441 and AIC-A-2011-0654.

P2-154 **Test of the Piezoceramic Motor Technology in ITER Relevant High Magnetic Fields**

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In the framework of a Fusion for Energy (F4E) grant, a new test campaign has been started in 2012 in order to fully exploit IVVS (In Vessel Viewing System) performances and verify its compatibility when exposed to ITER typical working conditions. Several tests simulating thermal vacuum environment, neutrons and gamma irradiations, and high magnetic fields that are equal or closely similar to the actual ones predicted in ITER are under way in ENEA laboratories. In this paper is reported the testing activity performed on a customized piezo-ceramic motor and on the main components of the actuating system of the IVVS probe with reference to magnetic field conditions. A customized motor, specifically produced by PI, has been adopted to study the performances of the device in high magnetic field conditions. The test was carried out in a test facility of ENEA laboratories able to achieve 14 Tesla. A maximum field of 10 Tesla, fully compliant with the ITER requirements (8 Tesla), was applied with a ramp rate of about 0.8 Tesla/minute. A specific mechanical assembly has been designed and fabricated to hold the motor and the associated sensors in the region of high homogeneity of the field, and a torque meter has been connected to the device to simulate a load. The effect on performances originating from unpredictable induced current has been taken into account by measuring the mechanical contribution alone, thanks to an auxiliary cc motor placed outside the field region and mutually exclusive with the piezo-ceramic motor. Results obtained so far indicate the compatibility of the motor to high magnetic fields, and are presented in the paper.

Topic J Plasma

P2-155 **Plasma current sustainement after iron core saturation in the STOR-M tokamak**

Mitarai, Osamu[1]; Ding, Yue[2]; Hubeny, Michael[2]; Lue, Yie[2]; Onchi, Takumi[2]; McColl, Dave[2]; Xiao, Chijin[2]; Hirose, Akira[2]

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As the space for a central solenoid (CS) is limited in a spherical tokamak (ST), it is desirable to start up the plasma current without CS. We propose to use of small iron core transformer to start up the plasma current in an ST reactor without CS. In this work, taking advantage of the high aspect ratio of the STOR-M tokamak, we have demonstrated that the plasma current up to 10~15 kA can be started up using the outer Ohmic heating coils without CS, and that the plasma current can be maintained after iron core saturation by further increasing the outer Ohmic heating coil current.

Experiments have been carried out by removing the inner OH coils and the normal vertical field coils in STOR-M. When the magnetizing current reaches 1.2 kA and the iron core becomes nearly saturated, the third capacitor bank is discharged to maintain the plasma current by an air core transformer. As a result the plasma current is slightly increased and maintained for 5 ms. It is clearly seen that iron core saturates on the hysteresis curve and air core transformer takes over the discharge.

The plasma position is controlled by feedback control vertical field coils placed on the vacuum chamber because the plasma position tends to move inside since the outer Ohmic heating coil produces a vertical field.

This is the first proof of principle experiment to demonstrate transition from the iron core operation without CS to the air core transformer operation. The results suggest that a small plasma current is produced during the iron core phase and a larger plasma current can be further ramped up by application of additional heating power and vertical field after iron core saturation in a future ST without CS.

P2-157 **Consequences of plasma disruption mitigation by massive gas injection on the ITER torus cryopumping system**

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The ITER torus exhaust pumping system is based on six primary vacuum cryogenic pumps to process the high gas throughputs during plasma operations, and a rough pumping system to pump the torus down to the cross-over pressure at which the cryopumps can be switched on. Roughing pumps are also used as forepumps needed for the regeneration of the cryopumps. Massive gas injection in the torus is a proposed concept to mitigate plasma disruptions but more investigations are needed to find the best candidate systems and gases. The objective of this work is to identify the limitations and potential down-times of the ITER torus pumping system after massive gas injection. Helium, deuterium, neon, and argon injections are considered. Gas flow calculations from the divertor dome towards the front of the cryopumps are discussed. In a second step, the resulting heat loads to the torus cryopumps and their temperature evolution are assessed and it is investigated to what extent the pumps go into regeneration. The pressure jump and high mass flows in the vacuum system are accompanied by large heat fluxes to the cryopumps, resulting in a significant and rapid increase of the pumping surface temperatures normally cooled at 5 K. However, the consequences on the cryopump operation depend on the gas species and quantities injected. Helium and deuterium injections are leading to the spontaneous regeneration of the cryopumps before pumping down the torus to the required pressure, and the cryopumps have then to be closed and regenerated. On the other hand, in the case of neon and argon injection, the torus can be pumped down without leading to the spontaneous regeneration of the cryopumps, thus reducing the down-time of the torus pumping system until vacuum and thermal conditions for the start of the next plasma pulse are initiated again.

P2-158 **Numerical simulation of ELMy H-mode in EAST using SOLPS**

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High-confinement mode (H-mode) in company with edge localized modes (ELMs) is an extremely important operation regime for a steady-state operation machine, since it can maintain the capability of tokamak confinement through periodically expelling impurities and energy from core plasma. ITER [1] is foreseen to operate in this regime. However, since the ELMs are difficult to control, divertor plates are very prone to be damaged by frequent giant ELMs. As a long-pulse superconducting tokamak device, EAST [2] has achieved ELMy H-mode in different power injection scenarios, i.e., with LHCD alone, ICRF alone, and combined LHCD and ICRH heating. Up to date, few studies have yet been conducted on modeling ELMy H-mode in EAST. These studies are very important and necessary for deeper understanding and better controlling ELMs.

The fluid code SOLPS5.0 [3] is employed in this work to study the time-dependent ELMy H-mode discharges in EAST. We can adjust the perpendicular anomalous transport coefficients (PATC) [4] by matching the experimental upstream profile under given H-mode discharge parameters to obtain the steady-state ELMy H-mode. In the modeling, the radial transport region is divided into three regions--inner core region, H-mode pedestal (edge transport barrier) and outer Scrape-Off Layer (SOL)---to accommodate the difference in transport rates between the edge pedestal region and the main SOL region. The divertor target profile extracted from the simulation is in good agreement with that obtained from Langmuir Probe and fast IR camera measurements during the corresponding EAST discharges. Furthermore, different types of ELMs are also modeled by periodically enhancing PATC with the parameters such as the repetition frequency and the energy expelled from core plasma, taken from the experimental data of EAST. The effects of these ELMs on the divertor plates are assessed.

References

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[3] R. Schneider et.al., Contrib. Plasma. Phys. 46 (2006) 3. [4] B. Gulejova et al., J. Nucl. Mater. 390-391 (2009) 412.

P2-159 **Conceptual design of dc power system for superconducting magnet of helical DEMO reactor FFHR-d1**

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The force-free helical reactor (FFHR) is a series of helical-type fusion reactor whose design is being studied at the National Institute for Fusion Science. FFHR is suited to steady-state operation and has the advantage that the magnetomitive force acting on the helical coil is balanced and it is not a experimental device, and the magnetic field configuration will be optimized for burning plasma.

This paper introduces a conceptual design for a dc power system to excite the superconducting coils of the FFHR.

The following conditions and requirements apply to the dc power supply:

1. The magnetomotive force should be balanced in any time.

2. When the fusion plasma is heated, the plasma axis moves as the heating progresses and this movement must be canceled by controlling the magnetic field.

For the first requirement, the operating currents of the superconducting coils are set to the same value, with selection of the turn numbers of the coils. With this setting, all the coils can be connected in series and the magnetomotive force are balanced in any time. For the next requirement, we add the control coils for the poloidal coils which is used in the ignition process to control the magnetic axis. The control coils were also set to the same operating current and connected in series. With this configuration, the power system is divided in two part. The one is a main power supply that has large current but low voltage rating, and the other is the additional control power supply that rating is higher voltage but low output current to sweep the magnetic axis. And the total capacity of the power supplies are decreased using this configuration.

P2-160 **Upgrade of the IR thermography diagnostic for the WEST project**

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To operate long plasma discharge tokamak, equipped with actively cooled plasma facing components (PFC), infrared (IR) thermography is a key diagnostic. Indeed IR data are used for both PFC safety monitoring, to avoid material degradation and water leak, and various physics studies on plasma-wall interaction. The IR monitoring is becoming even more crucial with today metallic armours, due to the risk of melting under abnormal heat flux whereas carbon based materials were able to handle it by sublimation. This is the case for the WEST project (Tungsten Environment for Steady State Tokamak) [1], which aims at installing a W divertor in Tore Supra (TS), in order to operate the 1st tokamak with a full W actively cooled divertor in long plasma discharged.

In the frame of the WEST project preliminary studies, the IR thermography system of TS is foreseen to be upgraded. It will consist of a set of 3 different diagnostics: 1/ Six cameras located in upper ports viewing the full W divertor, which reuse a part of the existing diagnostic (optical lines, middle IR waves cameras [3-5µm]). 2/ Five novel views located in the inner wall of the tokamak for the antennas monitoring, based on an innovative imaging fibres bundle technology associated with short IR waves cameras [0.9-17.µm]. 3/ An ITER like tangential wide angle view located in a median port, for the upper divertor and first wall monitoring.

The preliminary design of these 3 diagnostics, including photonic simulations, and the current development status, are described in this paper. A prototyping for the imaging fibres bundle associated with a complete simulation of instrument are undergoing to assess the feasibility to achieve measurements requirements. Performances evaluation in terms of spatial coverage and resolution of tangential and upper views are also reported.

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P2-161 **Neutronic Analysis for design of ITER IR thermography**

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Neutronic analysis is necessary for optimized design of diagnostics in ITER. In this paper, results of neutronic analysis applied to the ITER infrared (IR) thermography, which will be procured by Japan Domestic Agency, are presented. Optical mirrors will be installed in the equatorial (EQ) port plug. The design of the optical system is very important from the viewpoint of not only measurement accuracy but also neutron shield, because nuclear heating of optical mirrors and peripheral components could affect measurement accuracy as well as safety operations of the system. It is a target that a contribution of each diagnostic system installed in EQ port to shutdown dose rate (SDDR) in the inter-space region is up to 15 μSv/h in order that total SDDR is suppressed to be less than the limit value (100 μSv/h) for manual maintenance in ITER. In the neutronic analysis using MCNP version 5, the contribution of the SDDR in the interspace for the present optical system was evaluated ~ 20 μSv/h. Then, a dog-leg structure was added to the optical system to reduce streaming neutrons, while the measurement capabilities were maintained. As a result, the contribution were reduced to ~ 10 μSv/h for the new design. Infrared radiation due to nuclear heating in peripheral radiation shields of the optical system increases noise for measurements of IR light emission from the divertor plate. Therefore, nuclear heating of the radiation shield should be suppressed as low as possible. It is found by performing neutronic analysis that nuclear heating of peripheral components of the plasma facing mirror becomes half by reducing the diameter of the aperture of optical path between the plasma facing mirror and the plasma to half. Futher, materials of radiation shield and those combinations are now being optimized in order to reduce infrared radiation.

P2-162 **Matrix Converter Design for Feedback Stabilization of Vertical Position Instability on QUEST**

Nakamura, Kazuo[1]; Fujita, Hiroki[2]; Liu, Xiaolong[3]; Xue, Erbing[1]; Mitarai, Osamu[4]; Hasegawa, Makoto[1]; Tokunaga, Kazutoshi[1]; Zushi, Hideki[1]; Hanada, Kazuaki[1]; Fujisawa, Akihide[1]; Matsuoka, Keisuke[1]; Idei, Hiroshi[1]; Nagashima, Yoshihiko[1]; Kawasaki, Shoji[1]; Nakashima, Hisatoshi[1]; Higashijima, Aki[1]

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Fast-response power supply is indispensable for feedback stabilization of vertical position instability of highly-elongated spherical tokamak plasma, where decay index of the vertical magnetic field for equilibrium is negative. In the tokamak-type fusion reactor, higher transformation efficiency, less higher-harmonics and higher power factor are requested in addition to the high-responsibility. Three-phase to single-phase matrix converter is discussed for application to the feedback control of the horizontal field coil current (single-phase inductive load). In Alesina-Venturini method, the output voltage frequency is set to zero, and the voltage gain is modulated according to the external control command. Namely, the arbitrary output voltage waveform and the input displacement factor of unity are attained. In case of space vector modulation method, only the rectifier stage is adopted in virtual matrix converter and the arbitrary output voltage waveform is obtained by changing the modulation index. The input displacement factor of unity is attained by switching so that the input current is in phase to the input voltage. Though eddy current around the tokamak plasma decelerates the growing speed of vertical position instability, it may shield the stabilizing horizontal magnetic field into the stabilizing shell. So, high-voltage output is necessary for the forcing in addition to the above mentioned characteristics. As for the application of the matrix converter, the number of commutations should be optimized for the high voltage and high efficiency. First, the above optimization is discussed in alpha-beta-zero coordinate and direct-inverse-zero coordinate system, assuming the switching frequency is much higher than the modulation frequency. Second, it is discussed in detail by duty-cycle space vector approach. Third, design, fabrication and test are made concretely about a matrix converter: the switching frequency is 10kHz, the rated input phase voltage and current are 700V and 2.5kA, and the rated output ones are 350V and 2.5kA.

P2-163 **Projection of foreseeable integrated plasma performance to DEMO**

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Several concepts of DEMO have been proposed so far with plasma physics assumptions. At the same time, plasma performances foreseen in DEMO have been developed experimentally in tokamaks. However there are large gaps between the physics design parameters of the DEMO concepts and the simultaneous achieved parameters in tokamak experiments. Purposes of this paper are to clarify physics design issues and to project the foreseeable integrated plasma performances to DEMO in order to establish a feasible DEMO conceptual design.

Comparison of the achieved integrated plasma performance and the DEMO design parameters shows that the large gaps exist mainly in normalized electron density by the Greenwald density limit, normalized beta and radiation fraction in the exhausted power. The reason comes from trade-off relationships between plasma performances. The first design issue is the trade-off between the normalized electron density and the confinement enhancement factor. The experiments show degradation of the confinement with increasing the density by large amount of gas puffing, which leads to the decrease in fusion output. The second issue is the trade-off between the normalized beta and the confinement, because the reduced transport produces the steep pressure gradient while that reduces the beta limit, which leads to the decrease in fusion power density. The third issue is the trade-off between the radiation fraction and fuel purity. Although enhancement of radiation fraction by impurity seeding is necessary for reducing heat load on plasma facing components, penetration of impurity into the core plasma degrades the fuel purity, which leads to the decrease in fusion output. The DEMO physics parameters should be selected in consideration of the trade-off relationships observed in the experiments. Based on the foreseeable integrated plasma performance, the possible DEMO design parameters will be analyzed by using the system code TPC.

POSTER SESSION 3
Thursday 19th September
16:00- 18:00 Ground floor

Topic A First Wall

P3-001 **Hydrogen Retention in Plasma Facing Materials: The Influence of Material Microestructure**

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Pastor, Jose Ygnacio[4]; Perlado Martin, Jose Manuel[1]; Gonzalez Arrabal, Raquel[1]

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One of the challenges in the design of the future nuclear power plant is to develop materials capable to resist the hostile environment of a fusion reactor. Nowadays, tungsten is one of the most attractive materials proposed as plasma facing material (PFM) in nuclear fusion reactors. However, some limitations have been identified that have to be defeated in order to fulfill specifications i. e the light species retention. Light species (mainly H, D, T and He) are implanted in PFM, forming overpressurized bubbles and notably degrading W properties. In this work we focus on the study of H depth profiling in nanoestructured W (from now on NW) coatings as compared to the massive (from now on MW) counterpart. For this purpose resonant nuclear reaction (RNRA) experiments are carried out in NW and MW samples implanted with (i) H at an energy of 170 keV, (ii) sequentially implanted with C (665 keV) and H (170 keV) and co-implanted with C (665 keV) and H (170 keV). Implantations were carried out at a fluence of 5x1016 cm-2 and at two different temperatures RT and 400ºC. RNRA data evidence that the H concentration for samples implanted only with H is higher for NW than for MW, and it becomes comparable for both kind of samples after sequential implantation with C and H. Increasing the temperature during irradiation up to 400ºC leads H to completely release for NW as well as, for MW samples. The role of microstructure and radiation-induced damage on light species behavior is discussed.

P3-002 **Study of experimental simulations for the closed divertor using divertor simulator TPD-Sheet IV**

Kobayashi, Hiroaki[1]; Tanaka, Suguru[1]; Iijima, Takaaki[1]; Tonegawa, Akira[1]; Kawamura, Kazutaka[1]; Sato, Kohnosuke[2]

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2. Chubu Electric Power Co., Inc.

Closed divertor design for stable detached plasma formation should be optimized to handle high heat and particle fluxes. Design studies on closed divertor for divertor plasma formation are not easily understood, because the three-dimensional geometry of targets in the divertor plasma of tokamaks is complex. Thus, further study on the effect of the closed divertor configuration on detached plasma formation requires that the shape of divertor plasma, such as slab or sheet plasmas, be taken into account, in addition to variation of the target configuration parameters (e.g., length and angle).

Experimental simulations of both the effects of the leg length of the closed divertor on the formation of detached hydrogen sheet plasma in a linear divertor plasma simulator, TPD-Sheet IV [1], are presented in this report. TPD-SheetIV was constructed to investigate edge plasma physics in fusion research. The effect of the divertor leg length of the V-shaped target on the closed divertor is investigated with changing the contact gas flow rate. The divertor leg length of the V-shaped target is used for the accumulation of neutrals particles near the target plate. Measurements of the electron density ne, and the electron temperature Te, were conducted in a hydrogen detached plasma with a hydrogen gas puff. The emission intensities of the Balmer series spectra were observed using a spectrometer and a photomultiplier. The ionization and recombination events are discussed using the collisional-radiative (CR) model.

[1] A.Tonegawa, J.Nucl.Mater.,313-316 (2003)1046.

P3-003 **Deflection of a liquid metal jet in a tokamak environment**

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Employment of an array of liquid metal jets is envisioned as a means for exhausting the heat load generated in fusion reactors while protecting the plasma facing components. Liquid gallium is a plausible candidate for this application due to its relatively small vapour pressure and small chemical reactivity. The interaction of a liquid gallium jet with plasma has been investigated in the ISTOK tokamak. The jet was observed to remain stable during its interaction with plasma without any significant contamination of the latter during transient or steady operation. However, a significant deflection of the jet was detected as soon as plasma production was started. In addition, a strong dependency of the deflection magnitude on plasma position was observed that could be correlated with plasma potential gradients.

As a means to capture and, possibly, quantify this effect, a preliminary magnetohydrodynamic analysis was performed in order to predict the trajectory of a jet that is travelling inside an electromagnetic field. The effect of Lorentz forces, gravity and pressure drop are accounted for in a unidirectional model that assumes a small jet radius in comparison with the trajectory length. At steady state the pressure drop is determined by the relative importance of capillary and electric stresses at the jet/plasma interface. The effect of external electric potential gradients on jet deflection was ascertained in conjunction with the importance of electric stresses in accelerating the jet and reducing its radius. Stability analysis is required in order to identify the parameter range for which electric stresses will overcome capillarity and pinch-off droplets, as already reported in the ISTOK experiments. Possible asymmetries in the jet shape may be used as controlling parameters of jet deflection in order to facilitate future applications of power exhaust in fusion reactors.

P3-005 **Operational impact on the JET ITER-like wall in-vessel components**

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The JET ITER-like Wall (ILW) provides the same plasma facing component configuration as ITER during its active phase: beryllium in the main chamber and tungsten in the divertor. The paper will discuss the impact of the first 11 months of operation on the ILW plasma facing components. Emergency shutdown of a JET pulse with the carbon wall involved switching off the gas. With the ITER-like wall the low outgassing plus dynamic pumping meant that this same procedure produced extremely low plasma densities leading to slide-away electron beams which produced some very localised damage to beryllium limiters and main chamber tungsten coatings. This problem was eliminated by modification of emergency shutdown procedure. Disruptions with the ILW do not produce significant beams of runaway electrons, but they are different from those with the carbon wall where most of the available energy was radiated. With the ILW low radiation from beryllium impurities means high plasma temperature during the current decay which therefore is slower and delivers more energy to the wall leading to Be melting at the top of the machine and higher electromechanical impulse. Routine use of disruption management strategies including massive gas injection have proven successful in mitigating the risks to the machine. The main chamber power handling has achieved and possibly exceeded the design targets with effective shadowing of all edges over a range of magnetic configurations. Experiments designed to approach the design limits unintentionally caused toroidally non-uniform melting of the upper portion of some of the inner wall guard limiters and the outer wall poloidal limiters. The bulk tungsten divertor tile has been tested close to its energy and surface temperature limits without problems. There is no evidence of delamination due to thermal fatigue or significant erosion of W-coated divertor tiles in either the inner or outer divertor.

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*See appendix of F. Romanelli et al., Proc. of the 23rd IAEA Fusion Energy Conf. 2010, San Diego, USA

P3-007 **ITER EC H&CD Upper Launcher: Integration of Remote Handling features**

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Design concepts are presented of the integration of Remote Handling (RH) compatible components of the ITER ECH Upper Launcher (UL). The UL is currently in the Final Design stage. It is an in-vessel system that contains eight beamlines for transmission of 170 GHz mm-wave beams with a maximum power of 1.8 MW each, mounted into port plugs inside upper ports 12, 13, 15 and 16. No scheduled maintenance is foreseen for the UL port plugs, yet there remains a finite risk of component failure. Maintenance of the launchers shall be performed through the use of RH in the ITER Hot Cell Facility (HCF). Components for which new design and maintenance concepts are discussed in this paper are the steering mirror (M4), the Front Mirror (M3), the mid optics (M1, M2) and the waveguide assembly (WGA). The M3 and M4 mirrors are accessible by removing the Blanket Shield Module (BSM) that covers the plasma facing side of the UL. For replacement of the M3, cooling pipes are to be cut and re-welded, while the M4 has an additional pressurized Helium line, which is used for actuation. The proposed support structures offer good access for RH tools to all connections. The M1 and M2 mirrors, located roughly in the middle of the upper port plug, are mounted in rows of four (one mirror per mm-wave beam) onto blocks that are mounted into the side of the upper port plugs through apertures in the outer structure. These mirrors also contain cooling lines that need to be cut and restored. The proposed support structure enables the mirror blocks to be easily accessed and replaced from the side of the launcher. The WGA consists of a square flange containing all eight waveguide feed-throughs and tapered antennae, all mounted onto a rigid support frame that is directly attached to the flange. Once this support frame is extracted from the launcher inside the HCF, it offers access to each individual waveguide for replacement purposes. The paper will outline the typical procedure for replacement of a component, followed by a description of all features that are implemented into the design to facilitate the replacement process. Aspects like optical alignment, access and tooling requirements will be discussed.

P3-008 **Preliminary results on tungsten tile test in KSTAR**

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Plasma-surface interaction is one of critical issues in ITER. Recently, JET and ASDEX Upgrade have employed beryllium and full tungsten first walls for ITER-relevant wall conditions [1, 2]. Tore Supra has just started the 'WEST' project which includes major machine configuration change into a tungsten divertor machine [3]. Recent JET experiments with ITER-Like Wall (ILW) configuration (Be first wall and tungsten divertor) shows a remarkable step towards the success of fusion reactor, for instance, 1) a better wall condition for start up and the low impurity levels are observed, 2) the fuel retention is decreased by a factor of 10, and 3) carbon impurity is reduced by a factor of 20 while maintaining good confinement [4]. Still, the behavior of tungsten armor under ELMs (edge localized mode) during H-mode and long pulse operation has to be investigated. We have installed a bulk tungsten tile at a position of outer strike point. The tile has a castellation structure suggested by A. Litnovski et al. for ITER [5]. The tile is exposed for long-term exposure test during the entire KSTAR 2012 campaign. The surface morphology of the tile and the deposition inside the gaps are investigated. Although the tile was not severely damaged, some parts of surface were eroded and tungsten droplets were found. The deposition at the gaps of the castellation seems similar with that observed in DIII-D obtained by DIMES probe [5].

[1] G. F. Matthews et al., Phys. Scr. T128 (2007) 137

[2] O. Gruber et al., Nucl. Fusion 49 (2009) 115014

[3] E. Tsitrone, Private communication

[4] F. Romanelli, 'Overview of the JET Results with the ITER-like Wall', 24th Fusion Energy Conference, 8-13 October 2012, San Diego, USA

[5] A. Litnovski et al., ITPA Div/SOL meeting, Private communication

P3-009 **Measurement and calculations of long-lived radionuclide activity forming in the fast neutron field in ITER vacuum vessel composites**

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Measurement and calculations of long-lived radionuclide activity forming in the fission reactor fast neutron field were done, in some composites of ITER: 316 LN IG(vacuum vessel) and 316 L – the material of TF coil and PF coil jackets. The activation was conducted in June/July, 2012 in fast neutron irradiation facility of the MARIA research fission reactor. The steel samples were of 10x10x1 mm plate-shape and of approximately 0.8 g mass. The neutron flux density was measured by means of activation foil method and unfolding technique; thermal neutron fraction (up to 0.5eV), epithermal (0.5eV – 1keV) and fast (above 1keV) and was 0.11%, 11.76% and 88.13% respectively. The activation lasted 200 hours, the neutron flux density was 5.4x10E12 (2.88x10E12 above 500keV) n/cm2 s, whereas total fluence 2.53E18 cm-2. After the activation, the tested material was cooled for about 100 days. The activity measurement of the formed radionuclides was done by means of gamma-ray spectrometry method; because of significant activities of the samples point geometry was used with the distance source-detector 50 cm. The efficiency for this geometry was determined by means of MC method and verified by the measurement of point standard sources. The time of the measurement lasted about 20 hours, dead time not more than 13%, measurement uncertainties 1-5%. Activity calculations of the formed radionuclides were done by means of program FISPACT-2007 using the activation library EAF-2010 and experimentally determined neutron flux. Measured activity of long-lived gamma emitting radionuclides formed in the tested samples was 4.5 GBq/kg 100 days after activation; the dominant radio nuclides were Co-58 (about 80%) and Mn-54 (14.5%). The calculation to experimental value (C/E) differs for particular radionuclides and it is equal to 0.91-1.02 for Cr-51, 1.3-1.8 for Mn-54, Co-58 and Fe-59, 0.93-1.17 for Co-57 and 1.4-2.2 for Co-60. The results obtained give, on one hand, the information about the activity formed in composites of ITER in the fast neutrons field, and the radiation risk connected with that. On the other hand, this data will be useful during planned, in 2013, irradiations of these materials in the 14 MeV neutron field, in 6Li-D converter, being constructed in reactor MARIA in Poland. The results obtained will enable to conclude the influence of fast neutrons from the reactor, thus determine the undisturbed 14 MeV neutron outcome.

P3-010 **Comparison between FEM and High Heat Flux Thermal Fatigue Testing results of ITER Divertor Plasma Facing mock-ups**

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ITER divertor is an actively water cooled divertor (WCD); the Outer and Inner Vertical Target (OVT and IVT) are designed according to mono-block concept. The comparison between the hydraulic thermal-mechanical analysis by ANSYS 14 and the test results on small-scale mock-ups manufactured with the HRP technology was performed. During the thermal fatigue testing in the Efremov TSEFEY facility, only the surface temperature was measured. The Finite Element Analysis (FEA) is then important because allows knowing any other information (temperature inside the materials, local water temperature, local stress...). FEA was performed coupling the thermal-hydraulic analysis, that calculates the temperature distributions on the components and the heat transfer coefficient (HTC) between water and heat sink tube, with the mechanical analysis where the plasticity regime of the copper was considered. In particular the thermal analysis was performed using the CFD simulation by means of ANSYS CFX, neglecting the subcooled boiling regime. The comparison between analysis and testing results is based on the temperature maps of the loaded surface and on number of the cycles supported during the testing and those predicted by the mechanical analysis using the experimental fatigue curves for CuCrZr-IG, that is the structural material in the component. Also the behaviour for Cu-OFHC based on the experimental fatigue curves was considered and the ultimate tensile strength for W, because their failure affects the heat removal capability of the component.

P3-011 **Development of a Plasma Driven Permeation Experiment for TPE**

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The permeation of implanted tritium through first wall and divertor structures is expected to increase in importance in future long-pulse D-T fusion reactors, especially that associated with the expected higher temperatures of plasma facing components. Experiments on retention of hydrogen isotopes (including tritium) at temperatures less than 800 °C have been carried out in the Tritium Plasma Experiment (TPE) at Idaho National Laboratory[1][2]. To provide a direct measurement of plasma driven permeation in plasma facing materials at temperatures reaching 1000 °C, a new TPE membrane holder has been built to hold material membranes (≤ 1 mm in thickness) at high temperature while measuring tritium permeating through the material from the plasma facing side. This measurement is accomplished by employing a carrier gas that transports the permeating tritium from the backside of the membrane to ion chambers giving a direct measurement of the plasma driven tritium permeation rate. Isolation of the membrane cooling and sweep gases from TPE's vacuum chamber has been demonstrated by sealing tests performed up to 1000 °C of a membrane holder design that provides easy change out of membrane specimens between tests. Simulations of the helium carrier gas which transports tritium to the ion chamber indicate a very small pressure drop (~ 700 Pa) with good flow uniformity (at 1000 sccm). Integration of this permeation target holder into TPE operation and preliminary TPE testing are described in this paper. In addition, expected permeation rates are also discussed in detail.

[1] M. Shimada, R.D. Kolasinski, J. P. Sharpe, and R.A. Causey, Rev. Sci. Instrum. 82, 083503 (2011). [2] J. P. Sharpe, R.D. Kolasinski, M. Shimada, P. Calderoni, and R.A. Causey, J. Nucl. Mater. 390-391, 709 (2009).

P3-012 **Quality Evaluation of Hip Joints Using Ultrasonic Technique**

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The Hot Isostatic Pressing (HIP) process is a fusion welding method that allows to join both similar and dissimilar materials. First Wall Panels of the ITER blankets structure, composed of 316L Stainless Steel (SS), copper alloy (CuCrZr) and Berylium (Be) will be joined using this technique due to the complexity of the structure, material properties and the functional requirements of the blankets. According to ITER recommendations, ultrasonic examination shall be carried out in order to evaluate these joints. This work is focused on the evaluation of the quality of joints manufactured in different HIP conditions using ultrasonic technique in order to obtain the optimum parameters to be used in this welding process. For this purpose, automatic immersion technique and pulse-echo system have been used to obtain ultrasonic images. After identifying echoes corresponding to the bond, grain echo reference value has been established, considering the signal above this reference as possible not-strong bond. A colour scale has been defined to perform an easy-qualitative evaluation of each bond. CuCrZr/Be, Be/SS and Be/Cu bonds have been tested and comparison between them is presented. Results on the inspection show the influence of the coating used in the Be/CuCrZr bond in comparison with the materials jointed without any coating, as well as the repeatability of these tests. However, more research is needed in terms of signal processing to perform a quantitative evaluation.

P3-013 **Modeling First Wall Mechanical Fracture due to Thermal Shocks Using X-FEM**

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The first wall of a nuclear fusion reactor is a plasma facing material and it has to withstand the thermal shocks due to irradiation from the nuclear fusion reaction. The energy after each explosion of the small DT fuel-pellets in Inertial Confinement Fusion Reactors is released in a pulse mode. In dry wall concepts of chamber, without or low-density gas protection, the first wall is composed by a thin armor of refractory material that protects the structural material from high temperatures and other atomistic effects. X-rays and ions carry one third of the explosion energy. That energy is deposited in the first microns of the armor during pulses of microseconds. The armor reaches high temperatures located near the surface, while the thermodynamic heat is released through the structural material in contact with the coolant. Thermo-mechanic transient loads have been calculated by mean of finite elements for a tungsten coated steels first wall. Our study will be based on irradiation conditions relevant to the demonstration chamber of HiPER project. We have simulated the behavior of the reaction chamber for different armor and structural material thicknesses under shock ignition targets of 150 MJ with a repetition rate of 10 Hz. Coolant temperatures between 300 K and 800 K will be applied on the external face of the structural material to observe its effect. Finally, the crack mechanical failure due to thermo-mechanic transient loads has been studied using the extended finite elements, X-FEM. This analysis aims at optimizing the selection of thicknesses in the design of the future inertial fusion reaction chambers. The methodology can also be used in the first wall of Magnetic Confinement Fusion Reactor, where the discharges of plasma generate thermal shocks in located areas of the first wall.

P3-016 **First wall design for compact tokamak – neutron source.**

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Design of the first wall for compact tokamak has been developed. The requirements were the following.
-Small thickness because of limited dimensions of compact tokamak.
-Average heat flux from plasma 1.5 MW/m2; local - up to 5 MW/m2.
-Steady-state operation.
-Usage of the first wall as vacuum boundary should be possible.
-Water cooling.
The designed first wall consists from two shells made from bronze and stainless steel accordingly. Bronze one has ridges and depressions. Water coolant flows between shells. Plasma facing materials (beryllium, tungsten) are connected to bronze shell through intermediate copper layer. Thermal and stress analyses were carried out. Different parameters and loads were used for calculations – heat fluxes, water pressure and temperature. Boundary conditions for stress analyses, etc. The results of calculations showed that
-first wall design can stand In the conditions of the load of outer atmosphere;
-maximum temperature of bronze shell and beryllium as plasma facing material were not more than 230 C and 330 C accordingly at the heat flux from plasma 5MW/m2 and water temperature 100 C;
-Maximum equivalent Mises stresses were ~ 300 MPa in bronze shell and ~ 150 MPa in beryllium/ In order to obtain more exact information about hydrodynamics, heat transfer coefficients and possibility and reliability of plasma facing materials connection to bronze shell some mock-ups were manufactured. Experimental investigations of these mock-ups are being planned.

P3-017 **Tracer Techniques in the Assessment of Erosion and Property Modification of Plasma-Facing Components**

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Processes of material erosion, migration and deposition are decisive for material lifetime and fuel inventory. They may lead to a drastic modification of plasma- facing components (PFC). Experimental approach to determination of erosion phenomena involves the combined application of spectroscopy, surface probes for ex-situ studies and tracer materials. The term “tracer” denotes species introduced on purpose to the plasma edge, either by injecting rare (e.g. WF6) or isotope-labeled (e.g. N-15) gases, by ablating tracer material using lasers or, by exposing marker tiles coated with well defined sandwich-type layers of heavy and light elements. The amount and distribution of eroded, transported and then deposited species is determined by means of several surface-sensitive methods. At the TEXTOR tokamak a programme dedicated to testing of PFC and studies of erosion processes by tracer techniques has been carried out.

The intention of this contribution is to present an overview of results regarding: (a) the retention of nitrogen, N-15 marker, used for cooling the plasma edge in operations with tungsten components; (b) oxygen impact, O-18 marker, in oxidative fuel removal methods and (c) tungsten mobility studies using hexafluoride, WF6. Results are summarized below.

(i) Over 10% of the injected nitrogen gas is retained in PFC.

(ii) The greatest surface concentration of W is found near the injection: up to 1x1018 W cm-2 . The retention of injected fluorine is small (2x1016 cm-2) showing that WF6 can be used in migration studies.

(iii) The deposit near the injection point contains (besides W) a mix of species including N-15 (3x1016 cm-2) and He (1x1017 cm-2). Helium, originating from glow discharge wall cleaning and He-beam diagnostics, has been identified for the first time in deposits. The presentation of surface studies will be accompanied by discussion of local and core spectroscopy data from those tracer experiments.

P3-020 **Defect Evolution In Tungsten Under Helium Irradiation: A Comparison With Experimental Results.**

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Tungsten is proposed as first wall material in future fusion reactors due to its good refractory properties, low activation rate and low tritium retention. A common feature is that the most detrimental situations will take place in pulsed ion irradiation mode, regardless the confinement method: inertial (laser) or magnetic. In the case of laser fusion with direct target (e.g., HiPER project), the intense helium pulses resulting from every target explosion and the subsequent He retention lead to swelling, pore formation, sample exfoliation and embrittlement beyond certain He dose threshold. However, the helium-tungsten interaction is not well understood due to the subtleties imposed by the exact temperature profile evolution, ion energy, pulse duration, existence of impurities and simultaneous irradiation with other species. OKMC (Object Kinetic Monte Carlo) is a useful technique to simulate the helium evolution in large temporal and spatial scales, as well as vacancy clustering associated to the helium irradiation. With OKMC, we have simulated W irradiation with He pulses, including the sudden temperature raise (ìs range) and subsequent drop induced by every pulse. Comparison with experimental data and with continuous low intensity (tens of nA/cm2) He irradiation allow us to describe the evolution of He in vacancy clusters, the defect profile and infer the differences in terms of threshold damage for both types of irradiation, pulsed and continuous.

P3-021 **Impact of repeated high heat loading on surface modification of tungsten materials.**

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Tungsten (W) is the primary candidate for use as plasma facing materials/components (PFM/PFC). PFM/PFC will be subjected to heavy thermal loads in the steady state or transient mode combined with high energy neutron irradiation that will cause serious material degradation. In the present work, repeated pulse high heat loading experiments have been performed in order to investigate the damage characteristics of tungsten materials caused by the repeated heat loading during ELM.

ITER grade W, W-1.0wt%La2O3 and K(0.003wt%) doped W were machined to the dimensions of 10 mm x 10 mm x 1mm, followed by mechanical and electro polishing. All of the polished specimens were placed on a water-cooled Cu block and subjected to high heat load experiments by an electron beam irradiation test simulator of the Research Institute for Applied Mechanics, Kyushu University. The experiments were conducted at the irradiation conditions; repeated irradiations of 2 second-irradiation and 7.5 second-rest with one cycle of 9.5 seconds for totally 200 times. The surface temperature of the samples changes from below 450oC to 1300oC by 2 second-irradiation. Before and after the irradiation, the specimen surfaces were examined by SEM. In addition, quantitative analyses about temperature profiles and elastic-plastic thermal stress have been carried out using FEA In the case of ITER grade W, the repeated irradiations of 20 times caused surface roughening in the intragranular. The surface roughening is due to plastic deformation caused by the thermal stresses due to temperature difference. The subsequent repeated irradiations of totally 200 times caused significant surface roughening, cracking in the intragranular and grain boundaries and surface exfoliation. This suggests that the repeated heat loading during ELM causes this kind of surface modification. Therefore, estimation of influence of the surface modification on erosion, exfoliation, lifetime, thermal property and hydrogen retention will be required. In addition, the profiles of thermal stress during and after the electron beam irradiation were evaluated by the comparison with the experimental and the results of the FEA analyses.

P3-023 **Effect of transient heating loads on beryllium**

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Beryllium will be used as a plasma facing material for ITER first wall. The primary reasons for the selection of beryllium as an armor material for the ITER first wall are its compatibility with plasma and high oxygen gettering characteristics. In ITER the edge-localized modes (ELMs) will result in large thermal transient loads on beryllium components of first wall. These transient loads cause rapid heating of beryllium surface and can result in some changes in surface and near-surface regions such as material loss, melting, cracking, evaporation and formation of beryllium dust It is expected that erosion of beryllium under transient plasma loads such as ELMs and disruptions will mainly determine a lifetime of ITER first wall. This paper presents the results of recent experiments with the Russian beryllium of TGP-56FW ITER grade on QSPA-Be plasma gun facility. The QSPA-Be plasma gun facility, a quasi-stationary plasma accelerator, provides hydrogen (or deuterium) plasma heat loads corresponding to ITER ELMs and disruptions in the range of 0.2-5 MJ/m2 and a pulse duration 0.5 ms. The Be/CuCrZr mock-ups were tested by deuterium plasma streams (5 cm in diameter) with pulse duration of 0.5 ms and average heat loads of 0.5 and 1 MJ/m2. The angle between plasma stream direction and target surfaces was 30î. Experiments were performed at ~250oC. The beryllium targets were exposed to up to 100 shots. After 10, 40 and 100 shots, the evolution of surface microstructure and cracks morphology were investigated as well as beryllium mass loss under erosion process. The study of erosion products in the form of beryllium dust particles and porous films will be also presented.

P3-024 **Development of Materials for the First Wall of a Nuclear Fusion Reactor**

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The purpose of the first wall in a nuclear fusion facility is to protect the underlying structural material against the severe irradiation conditions that will take place. In particular, in the case of laser fusion with direct targets (e.g., HiPER project), intense pulses of ions will arrive at the first wall after every explosion. Tungsten is the material of choice due to its good refractory properties, low sputtering and low tritium retention. However, even if a W first wall is designed to accommodate the ion pulses, the accumulation of light species and the formation of pressurized bubbles lead to fatal failure due to W exfoliation with mass loss. We have undertaken an important effort to develop new W-based materials with superior resistance to ion irradiation. By means of advanced sputtering methods we grow nano-columnar structures of W with good control over the grain size and grain orientation. Subsequently, we irradiate the grown samples with double or triple beams to study the effects of the light species or with plasma discharges to study the thermomechanical response of the samples. Characterization based on electron microscopy, nuclear techniques and mechanical tests allow us to draw a clear picture of the radiation effects on our samples. Furthermore, we have developed a complete methodology for simulation or irradiated materials based on multiscale modeling. Making use of data from DFT and MD simulations we have parametrized the W-light species system for Kinetic Monte Carlo simulations. By means of these simulations we can explain the evolution of defects from vacancies to pressurized bubbles. At a higher level, finite elements simulations simulate the thermo-mechanical behavior of the first wall. Finally, the fracture mechanics such as crack growth have been also studied using the novel methodology of extended finite elements.

In this paper we will present our joint efforts in sample growth, sample irradiation, characterization and simulation with regard to the development of new materials for first wall applications in laser fusion materials. The methodology presented here could be useful for other related fields.

P3-025 **Status of the IDTF high-heat-flux test facility**

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ITER Divertor Test Facility (IDTF) was created at the Efremov Institute in the frames of ITER program. The IDTF is intended for the high heat flux tests of the outer vertical targets, the inner vertical targets and the domes as for qualification of design solutions and potential manufacturers so for manufacturing quality control at the serial production of components. A heat flux is generated by an electron-beam system (EBS). Power of this EBS is 800 kW and a maximum accelerating voltage is 60 kV. The test assembly of the plasma facing units of the ITER divertor is mounting on the manipulator, which is designed for positioning and movement of a test object inside the vacuum chamber. Dimensions of the vacuum chamber permits to place inside test objects up to 2.5 m in length and 1.5 m in width. Vacuum pumping system provides the pressure inside the chamber about 3 10-5 mbar. Cooling system produces cooling of test objects, it circulates deionized water, the same parameters and water quality that are in the corresponding cooling system of the ITER reactor. Electric power consumed by the facility reaches 1.3 MW. Integrated control system manages all of the subsystems of the IDTF and data acquisition from all diagnostic devices, such as pyrometers, IR-cameras, video cameras, flow, pressure and temperature sensors. The IDTF was creating from 2008 up to 2012 and in summer 2012 had been qualified for the testing of the outer vertical full-scale prototypes. The technical description of test facility is presented in the report. At the end of 2012 the high heat flux testing of the test assembly of the outer vertical target full-scale prototype was completed. The testing procedure is presented in the report.

P3-026 **Computational Methodology For Study Nuclear Fusion Materials And Systems**

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A complete computational methodology has been developed in the "Instituto de Fusión Nuclear" to study materials and systems in the field of nuclear fusion by Inertial Confinement (IC) and Magnetic Confinement (MC). Materials and systems of future nuclear fusion reactors have to withstand the harsh environment that involves the radiation of high energetic particles. Computational methodologies with specific codes are being used to solve the complex problems coming from materials development to reactor design. Starting with the lower scale, ab-initio simulations are carried out to study the hydrogen solid phase that forms part of the external layer of targets used in a IC reactor. Molecular Dynamic calculations provide the atomic damage induced by radiation in materials for future IC and MC reactors. Interstitial defects stability in FeCr alloys, dislocation mobility in Tungsten, amorphization of SiO2 due to ion implantation and the behavior of liquid Lead-Lithium have been studied using Molecular Dynamics. In a higher level, Kinetic Monte Carlo is used to simulate the defect evolution into the first wall in an IC device during the pulsed operational mode. Finite elements simulations provide the thermo-mechanical behavior of materials irradiated with high energy particles, such as the first wall and final optics in IC reactors. Making use of the novel methodology of extended finite elements, these methods provide the fracture mechanics due to crack growth. In addition to the simulation of materials under irradiation we are carrying out engineering studies of fusion power plant systems. Transport codes are being used for diffusion and retention of light species, such as tritium in the reactor chamber and for neutronic activation and radio protection of future IC nuclear reactors. The thermal-cycle of the power plant is being studied combining finite elements with computational fluido-dynamics and transport codes. Furthermore we are applying our methodology to other nuclear systems in order to specify design parameters. Remarkably, we are developing key systems for the European spallation facilities (ESS-Bilbao and ESS-Lund)

Topic B Blankets

P3-027 **Numerical study of the MHD flow characteristics in a three-surface-multi-layered channel with different inlet conditions**

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MHD pressure drop reduction is one of the important R&D issues in the self-cooled Li/V blanket. A three-surface-multi-layered channel, which consists of a thin metal layer and a insulating layer on three of the four channel walls, is an advanced concept to reduce the MHD pressure drop. In order to validate this concept, a liquid metal flow experiment has been conducted by using the three-surface-multi-layered channel. The experimental result was slightly larger than the two-dimensional numerical one. In this study, the difference between these results was examined by using three-dimensional MHD flow simulation. Since the pressure drop was larger in the upstream region than the one in the downstream region, an effect of the inlet flow condition to the test section was discussed. The governing equations were Navier-Stokes equations, the continuity equation and a Poisson equation of the electrostatic potential. A finite volume method with a collocated grid system was employed for this simulation to calculate the current density with high accuracy. The pressure field was obtained by a SIMPLE-type numerical scheme. A test section consisted of the test channel with three-surface-multi-layered structure and corn-shaped connectors to connect the rectangular test channel and the circular pipe of the experimental loop. In this simulation, the connectors were replaced by simple rectangular conducting channels. In the case of the inlet condition with an MHD flow profile in a simple conducting channel, the pressure drop declined in the upstream region in the test channel relative to the one in the downstream region. This is an opposite characteristic to the experimental results. On the other hand, in the case with a turbulent flow profile as the inlet condition, the decrease in the pressure drop did not occur. These results meant that the inlet condition affected the pressure drop characteristics in the test section.

P3-028 **Fabrication and characteristics of SiC-coated graphite pebbles for HCCR TBM**

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The Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM) uses graphite in the form of pebbles as a neutron reflector, opening the possibility of reducing the amount of beryllium neutron multiplier. A silicon carbide (SiC) coating should be formed on the graphite pebbles to prohibit the reaction of graphite with steam and/or air during accidents and to reinforce the surface strength. In this work, a dense ¥â-SiC coating was formed over the surface of the graphite pebbles by a chemical vapor reaction (CVR) process. CVR process is based on carbothermal reduction, in which silica reduction is carried out with carbon source. The microstructural features, crystalline structure, pore size distribution, porosity and oxidation behavior of the SiC-coated graphite pebbles were investigated. A SiC coating with a thickness of approximately 30 § remarkably improved the oxidation resistance and the density of the graphite pebbles. Before coating, the graphite pebbles showed an open porosity of 16.9%; after coating, the open porosity of the graphite pebbles decreased markedly to 12.9%. In an isothermal oxidation test conducted at 700¡É in air, the SiC-coated graphite pebbles showed strong oxidation resistance and the weight loss of approximately 2 wt% over the course of 2 hrs. New important results obtained from our experiments on the CVR-SiC coating of nuclear graphite pebbles are also introduced in this paper.

P3-029 **Evaluation of the Response Time of H-Concentration Probes for Tritium Sensors**

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Dynamic tritium concentration measurement in lithium-lead eutectic (Pb-15.7(2)Li) is of major interest for a reliable tritium testing program in ITER TBM and for an experimental proof of tritium self-sufficiency in liquid metal breeding systems.
Potentiometric hydrogen sensors using different solid-state electrolytes for molten lithium-lead eutectic have been reported and tested by the Electrochemical Methods Lab at Institut Quimic de Sarria (IQS) at Barcelona. The goal of the first set of experiments was to compare the electrochemical measurements obtained at a fixed hydrogen partial pressure at the working and the reference electrode with the theoretical potential calculated using the Nernst equation. Finally, the most promising elements have been selected for a second set of experiments in order to evaluate the sensor response time at different working temperatures.

In the present work the following ceramic elements have been synthesized and characterized by X-Ray Diffraction (XRD) in order to be tested as PEM H-probes: BaCeO3, BaCe0.6Zr0.3Y0.1O3-d and Sr(Ce0.9-Zr0.1)0.95Yb0.05O3-d. Potentiometric measurements of the synthesized ceramic elements have been performed shifting from a fixed hydrogen partial pressure at the working electrode to high purity argon. In this experimental campaign a fixed and known hydrogen pressure has been used in the reference electrode. The goal of these experiments is to evaluate the sensor response time when the hydrogen concentration in the environment is rapidly changed. All experiments have been done at 500 ºC and 600 ºC.

P3-030 **Development of a Hydrogen Permeation Sensor for Future Tritium Applications**

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Tritium monitoring in lithium-lead eutectic (Pb-15.7Li) is of great importance for the performance of liquid blankets in fusion reactors. In addition, tritium measurements will be required in order to proof tritium self-sufficiency in liquid metal breeding systems. On-line hydrogen (isotopes) sensors must be design and tested in order to accomplish these goals. The Electrochemical Methods Lab at Institut Quimic de Sarria (IQS) has been working under the Consolider Ingenio 2010 Program (sponsored by the Spanish Ministry of Science and Innovation) in the development of potentiometric hydrogen sensors using solid-state electrolytes for molten lithium-lead eutectic. In the present work a new approach has been addressed, the use of hydrogen permeation through palladium and α iron in order to build and finally test a permeation hydrogen sensor. In this work, an experimental set up was designed in order to test the permeation hydrogen sensors at 500ºC. This experimental set-up allowed working with controlled environments (different hydrogen partial pressures) and the temperature was measured using a thermocouple connected to a temperature controller that regulated an electrical heater. In a first set of experiments, a hydrogen sensor was constructed using an α iron capsule as an active hydrogen area. The sensor was mounted and tested in the experimental set up. In a second set of experiments the α iron capsule was replaced by a welded palladium thin disc in order to minimize the death volume. Like in the previous situation, the sensor was mounted and tested in the experimental set up. In both cases, the sensors response time were measured at different hydrogen partial pressures.

P3-031 **Current Status of Accident Analysis for Korean HCCR TBS**

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Recently Korea has decided to test Helium Cooled Ceramic Reflector (HCCR) Test Blanket Module (TBM) in ITER and design of the TBM with its ancillary systems, i.e. Test Blanket System (TBS), is under progress. Since the TBM is operated at elevated temperature with high heat load, safety consideration is essential in design procedure. In this paper, preliminary accident analysis results for the current HCCR TBS design on selected scenarios are presented as an important part of safety assessments. To simulate transient thermo-hydraulic behavior, GAMMA-FR code which has been developed in Korea for fusion applications was used. The main cooling and tritium extraction circuit systems, as well as the TBM, were simulated and the main components in the TBS were modeled as the associated heat structures. The important accident scenarios were produced and summarized in the paper considering the HCCR TBS design and ITER conditions, which cover in-vessel Loss Of Coolant Accident (LOCA), in-box LOCA, ex-vessel LOCA, Loss Of Flow Accident (LOFA) and Loss Of heat Sink Accident (LOSA). The accident analysis based on the selected scenarios was performed and it was found that the current design of the HCCR TBS meets the thermo-hydraulic safety requirements while it is envisaged to need more precise model for some components such as circulator, cooler, etc for future works.

P3-032 **Analysis of Electromagnetic Loads on EU-DEMO Inboard and Outboard Blanket Vertical Segments**

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Plasma disruptions and vertical displacement events (VDEs) in tokamak reactors are design drivers for the in-vessel components' attachments as the induced loads constitute a severe issue for the mechanical structure. The analysis of the electromagnetic (EM) forces acting on a model of fusion reactor is presented in this work as part of the ongoing EU DEMO studies. The DEMO1 configuration, elaborated in 2011 by EFDA on the basis of PROCESS system code optimization is here considered. It consists of a 3D model of the reactor designed on the assumption of a Multi Module Segment (MMS) structure, with vertical ports for inserting and removing of the MMSs. Each MMS consists of a number of modules which are connected to a strong manifold structure. This paper presents the time behaviour of EM loads acting on the inboard and outboard blanket segments due to a major central plasma disruption with a linear quench of 42 ms. The FEM model represents a 22.5-degree sector of the reactor. It takes into account for all the important components needed for the estimation of the EM loads on the blanket segments. Including also the toroidal field coils in the model allowed the calculation of a more realistic magnetic field distribution than assuming the analytical 1/r distribution of the toroidal field. The inclusion of non-linear B-H curves for the ferromagnetic materials together with the description of the total magnetic field and its saturation effects, results in a new state-of-the-art complexity level. EM analyses have been performed using the commercial ANSYS© code and considering both the HCPB and HCLL concepts for the breeding zone of the modules. The EM loads have been calculated for several components. In general, the Lorentz's forces (moments) are affected by the neighboring components and are in the range 0.08-0.4 MN (0.3-4.2 MN m) for the single modules, while they rise to 0.5-2 MN (20-46 MN m) when the entire blankets are considered. The results are also compared to linear analyses to underline the effect of the non-linearity of the ferromagnetic material.

P3-033 **On the numerical assessment of the thermo-mechanical performances of the DEMO Helium-Cooled Pebble Bed breeding blanket module**

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Within the framework of the European DEMO Breeder Blanket Programme, a research campaign has been launched by the University of Palermo, ENEA-Brasimone and Karlsruhe Institute of Technology with the specific aim to theoretically investigate the thermo-mechanical behaviour of the Helium-Cooled Pebble Bed (HCPB) breeding blanket module. Attention has been focussed on the HCPB/1 design option envisaged for the Helium-cooled DEMO1 blanket vertical segment, in order to investigate its thermo-mechanical performances under the Normal operation and Over-pressurization loading scenarios.

The research campaign has been carried out following a theoretical-computational approach based on the Finite Element Method (FEM) and adopting a qualified commercial FEM code. A realistic 3D FEM model of the central poloidal-radial region of the HCPB blanket module has been developed including one breeder cell in the toroidal direction and all the five cells in the poloidal one. No Breeder Units (BU) have been modelled, their presence being simulated by imposing effective thermo-mechanical loads. Two set of uncoupled steady state thermo-mechanical analysis have been carried out with reference to the investigate loading scenarios, assuming proper thermal and mechanical loads and boundary conditions, partly resulting from the assessment of the module thermal-hydraulic performances carried in parallel by the Karlsruhe Institute of Technology adopting a computational fluid-dynamic approach. In particular, it has been assumed that under the Normal operation scenario (level A) the module undergoes both 8 MPa coolant pressure on its cooling channels walls and thermal deformations due to the flat-top plasma operational state thermal field, while under the Over-pressurization scenario (level D), induced by a coolant leak, the module undergoes 8 MPa coolant pressure on its internal walls while it operates at the normal operation thermal level. Results obtained are presented and critically discussed according to the SDC IC and RCC MRx codes.

P3-034 **Study on effect of moderating materials on Tritium Production Rate in IN-LLCB TBM**

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India is developing Lead Lithium cooled Ceramic Breeder (LLCB) TBM to be tested in ITER. Liquid Lead Lithium along with Lithium Titanate has been adopted as basic material in Indian TBM for neutron multiplication and tritium breeding. Reduced Activation Ferritic Martensitic Steel (RAFMS) is used as the structural material and the first wall is cooled by Helium. Li-6 enrichment is taken as 60 and 90 % in Lithium Titanate and Lead Lithium respectively. Neutron spectrum from fusion plasma is peaked at high energy whereas the Tritium breeding reaction Li-6(n, α)H-3 has high neutron absorption cross-section at lower energies. A separate study has been launched to assess the effect of neutron moderators on Tritium production capabilities of LLCB DEMO blanket. For the neutronic design of the LLCB TBM, a detailed 3-D neutronic model of LLCB TBM in equatorial port at ITER has been constructed. A 3-D neutron source has been used for the D-T neutrons emitted by plasma. Design of TBM has been carried out based on plate type geometry. An assessment study has been carried out to increase the TPR by introducing neutron moderators such as Zirconium Hydride/Graphite in LLCB TBM. Study has been carried out using Monte Carlo code with Fendl-2.1 library. This paper will present the main findings from this neutronic study.

P3-035 **Neutronics of LiPb blanket and design and evaluation of integral experiment with D-D neutrons**

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In order to evaluate tritium production capability, neutronics analyses have been conducted for DT and DD plasma operations using Monte Carlo Method. We have used Monte Carlo calculation code MCNP and FENDL-2.1 (Fusion Evaluated Nuclear Data Library). Four blanket concepts have been evaluated for the followings;(a) W armor, F82H first wall, SiC and He coolant panel, natural LiPb breeder (b) W armor, SiC first wall, SiC and He coolant panel, natural LiPb breeder, (c) W armor, F82H first wall, SiC and He coolant panel, natural LiPb breeder, Be multiplier, (d) W armor, F82H first wall, SiC and He coolant panel, enriched LiPb breeder. TBR_DT obtained by DT operation for the concept (a), (b), (c) and (d) are 0.655, 0.662, 0.846 and 1.469, respectively. TBR_DD obtained by DD operation are 0.444, 0.478, 0.531 and 1.024, respectively. High TBR with enriched 6Li is expectable rather than using Be as a neutron multiplication material. The integral experiment using D ion beam is planned for the designed blanket concept. D ion beam is expected as a DD neutron source for experiment. DD reaction between D ion beam and adsorbed D on a blanket module can generate neutron, and produce tritium in the module. From the calculation results, it has been clarified that tritium produced by the DD neutrons can be detected in a small LiPb-SiC-Be module and compared with the calculation results. We have proposed an innovative operation scenario of DT reactor without initial tritium loading. Using tritium produced from DD plasma operation in lithium lead blanket with self-sufficient TBR (Tritium Breeding Ratio), production and breeding of tritium to satisfy full DT operation is possible by reasonable period of DD operation. Assuming an extended DD operation that generates small amount of tritium, followed by the partial DT to take advantage of multiplication by TBR, will result in considerable amount of tritium.

P3-036 **Influence of chemisorption products of carbon dioxide on radiolysis of tritium breeding ceramic**

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Lithium orthosilicate pebbles with 2.5 wt% excess of silica are European Union's designed tritium breeder for fusion reactors [1]. On the surface of the pebbles accumulates tritium, due to the interaction with radiation defects and products of radiolysis [2]. This effect could be explained by the fact that a radiation unstable lithium carbonate phase forms in chemisorbing process of carbon dioxide [3]. The aim of the investigation was to estimate the influence of carbon dioxide and water vapour on the radiolysis of lithium orthosilicate ceramic.

Nano-strcutured lithium orthosilicate powder with excess of silica was selected as investigation material. The powder was thermally treated up to 573 K for 4 hour in air, to accumulate water vapour and carbon dioxide [4], and irradiated with gamma rays (absorbed dose 56 kGy). The composition of the powder was analyzed with p-XRD, FT-IR and TG-DTA, the accumulated radiation defects were studied with ESR and chemical scavenger method. It was established that X-ray amorphous chemisorption products of water vapour and carbon dioxide, i.e. lithium hydroxide and carbonate, accumulate on the surface of lithium orthosilicate powder after thermal treatment in air. Radiation defects of chemsorption products are unstable and accumulate only in low amounts. Radiolysis of chemisorption products essentially affects the radiation stability of lithium orthosilicate powder and increases the concentration of radiation defects up to 50%. On the basis of the obtained results it was concluded that a lithium carbonate containing layer on the surface of pebbles can reduce the radiation stability and may cause tritium accumulation. It may be favourable to change the ceramic composition and to replace silica by a less reactive and mechanically stable phase like titania [5].

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P3-037 **Gas Absorption and Discharge Behaviors of Lithium-lead**

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When one conducts the experiment of heat and mass transfer by using lithium-lead, which is the most promising candidate coolantfor DEMO reactor, inert gas, such as argon, is used as cover gas. In our previous thermo-fluid experiment by using lithium-lead, we found the strange phenomena. The cover gas pressure enclosed inthe very leak tight container of lithium-lead ingot was decreased with time, that is, the gas-absorption of the solid lithium-lead at room temperatureunder atmospheric pressure.If this is true, it might be expected the gas-discharge from the solid lithium-lead, vice versa. There is no report regardingthese phenomena until today.Therefore, it is necessary to investigate whether these phenomena exist, because such a phenomenonpossibly involves the permeation behavior of smaller molecular gas, such as hydrogen and helium.In the experiment, the solid lithium-lead was enclosed in a cylinder of which temperature was well controlled. The pipe was connected with the above cylinder by a valve. The argon gas were initially enclosed in the pipe at almost atmospheric pressure for the absorption test, while the pipe was evacuatedup to about 4x104Pa for the discharge test. The behaviors of absorption/discharge of argon was detected as the pressure change in the pipe by a leak sensor.The pressure was decreased instantly after the connection valve was openedeven in the case of lithium-lead ingot, which reveals the fact of inert gas absorption under atmospheric pressure and room temperature. The rate and amount of absorption was increased with increase in temperature of lithium-lead. Conversely, the pressure was increased in the case of evacuation and the discharged gas was confirmed as argon by Thermal Desorption Spectroscopy, which shows the fact of argon gas-discharge. The amount of discharged gas was about one-seventh of absorbed gas volume.

P3-038 **Numerical Simulation of Buoyancy Effects of MHD Flow for ITER SLL-TBM**

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The liquid lithium-lead(PbLi) breeder blanket concept is an attractive candidate for test in ITER. To check and validate the feasibility, the China dual-functional lithium lead test blanket module(DFLL-TBM) system is designated to demonstrate the integrated technologies of both He single coolant(SLL) and He-LiPb dual-coolant(DLL) blanket. In SLL blanket, the lithium lead is quasi-static and just considered as tritium breeder, the heat in the breeding region is cooled by the helium.

An important aspect of MHD flows in SLL-TBM is related to non-uniform volumetric heating by fusion neutrons causing buoyant flows, which may influence or even dominate the performance in SLL-TBM where the breeder/coolant forced flow is quasi-static. In these conditions, the buoyancy forces affect the velocity profiles significantly and create recirculating flows, whose magnitude of velocity can even overcome that of the forced flow. In this paper the main characteristics of buoyancy effects of MHD flows in poloidal duct in SLL-TBM are described numerically, and the knowledge of buoyant convection in the presence of an imposed magnetic field is studied to understand the liquid metal circulating in the blanket. Effects of the direction of the heat flux with respect to the orientation of the magnetic field and the influence of electric conductivity of walls on the flow structure are investigated. Of particular interest is the formation in boundary layers of velocity jets driven by temperature gradients.

P3-039 **Heat load test with the HCCR TBM first wall mock-up and the GAMMA-FR code validation**

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A scale downed helium supplying system (HeSS), linked with an electron beam heat load facility, has been constructed at Korea Atomic Energy Research Institute (KAER) to demonstrate the helium cooling system of the HCCR TBM and to obtain thermal-hydraulic experimental data for validation of the GAMMA-FR (Gas Multicomponent Mixture Analysis for Fusion Reactor) code. HeSS supplied temperature of 300 ^{o}C, pressure of 8 MPa and mass flow rate of up to 0.5 kg/s (1/3 of the HCS) helium gas flow into the first wall mock-up of HCCR, which was installed in the electron beam facility (KoHLT-EB) with constant heat flux (0.3 - 0.5 MW/m2). From the heat load tests with the first wall mock-up, thermal-hydraulic experimental data were obtained with various heat loads and helium mass rate conditions. In present study, the measured temperatures of the mock-up and of helium gas during the heat load tests are compared with the calculated results of the GAMMA-FR code for the code validation. As results, the calculated temperatures are well agreed with the measured one and base on the comparison results, the heat transfer model for the first wall of HCCR in GAMMA-FR is confirmed under ITER operation condition.

P3-040 **First-principles study of hydrogen adsorption and permeation in the reconstructed cubic erbium oxide surfaces**

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Erbium oxide is one of multi-functional metal oxides widely used in electronics. Increased attentions to erbium oxide coatings have been attracted in the application of tritium permeation barrier (TPB) in fusion blanket systems due to their excellent properties, such as high permeation reduction factor, good compatibility with liquid lithium, and high electrical resistivity. Understanding surface properties of erbium oxide, especially adsorption and penetration of hydrogen isotopes, is of considerable importance in the interest of TPB coatings. In this work, we studied cubic low-index (0 0 1), (1 1 0) and (1 1 1) surfaces of erbium oxide, and employed spin-polarized density-functional theory (DFT) calculations as implemented in the Vienna Ab Initio Simulation Package (VASP) and the Generalized Gradient Approximation of Perdew and Wang for electron exchange and correlation. The cut-off energy was 500.0 eV and K-point sampling of 2 × 2 × 1 was used for (1 × 1) unit cells, converging the residual forces on each atom less than 0.01 eV/Å. The surfaces were modeled by a slab periodically in all directions, with 12-16 stoichiometric layers relaxed and unrelaxed. The vacuum layer between the slabs is 14.0 Å found sufficient to isolate the erbium oxide slab from its periodic image. As a result, we explored the surface structural and electronic properties of erbium oxide and investigated absorption and penetration of hydrogen at the three cubic erbium oxide surfaces. The energetic of hydrogen penetration from the surfaces to the solution site in the bulk were defined. It was found that for our low surface coverage of hydrogen (0.89 × 1014 H/cm2), the penetration energy of at least 2.0 eV is required for all low-index erbium oxide surfaces, which will provide useful guidance for future studies on modeling defects (e.g. grain boundaries and vacancies) in TPB coatings.

P3-041 **Visualization Experiment of Complex Flow Field in a Sphere-packed Pipe by Detailed PIV measurement**

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A Flibe blanket which uses molten salt Flibe as a coolant as well as a tritium breeder is considered as one of the potential candidates of fusion blankets. Using Flibe has several advantages, such as low electrical conductivity causing low MHD pressure drop and so on, however, Flibe is a high Prandtl number fluid and it has poor heat transfer performance. Therefore the heat transfer performance should be strongly enhanced to use Flibe as a coolant in fusion blankets. Moreover, in the concept of the Flibe blanket, the first wall is supposed to be cooled also by Flibe, and the heat flux up to about 1 MW/m2 is assumed to be imposed on the first wall. A sphere-packed pipe (SPP) has been proposed as a heat transfer promoter for the first wall cooling using Flibe. Many researches were done to clarify the heat transfer characteristic by means of experiments, and good heat transfer performance of the SPP was shown. However, the flow field in the SPP has yet to be elucidated in detail. In this study, the SPP was scrutinized to clarify the complicated turbulent flow field by means of PIV measurement and what enhanced heat transfer in the SPP. The geometrically complicated system was visualized by matching refractive index of acrylic resin as the channel material and sodium iodide solution as the working fluid. The visualization experiment was conducted with Reynolds number based on the sphere diameter and Darcy velocity of 5,000. As a result, it was found that high velocity regions appeared in the vicinity of packed sphere near penetrating paths formed in the SPP. In addition, it was clarified that turbulence energy was mainly generated in the large velocity shear regions formed near the pipe wall behind the packed spheres.

P3-042 **Rational of Helium Cooled Pebble Bed Blanket and R&D Activities**

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This work aims to give an outline of the design requirements of the Helium Cooled Pebble Bed (HCPB) Blanket and its associated R&D activities. In DEMO fusion reactor the plasma facing components have to fulfill several requirements dictated by safety and process sustainability criteria. In particular the Blanket of a fusion reactor shall transfer the heat load coming from the plasma to the cooling system and also provide Tritium breeding for the fuel cycle of the machine. Since the 80-ties, KIT has investigated and developed a Helium-cooled Blanket based on the adoption of separated small lithium orthosilicate (Tritium breeder) and Beryllium (neutron multiplier) pebble beds, the HCPB blanket. One of the Test Blanket Module of ITER will be a HCPB type, which aims to demonstrate the soundness of the concept for the exploitation in future fusion power plants. A discussion is reported also on the development of the design criteria for the blanket to meet the requirements, such as Tritium environmental release, with reference also to the TBM. The selection of materials and components to be used in a unique environment as the Tokamak of a fusion reactor requires dedicated several R&D activities. For instance, the performance of the coolant and the Tritium self-sufficiency are key elements for the realization of the HCPB concept. Experimental campaigns have been conducted to select the material to be used inside the solid breeder Blanket and R&D activities have been carried out to support the design. The paper discusses also the program of future development for the realization of the HCPB concept, also focusing to the specific campaigns necessary to qualify the TBM for its implementation in the ITER machine.

P3-043 **Engineering analyses of ITER divertor Thomson scattering**

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The support structure for the divertor Thomson scattering equipment– the diagnostic rack, actually play plugging role of the divertor port, should be designed to sustain strong electromagnetic, thermal and seismic loads. Meeting the requirements of corresponding analyses (which are often contradicting each other) leads to the iterative design process. The number of design drawbacks was performed in 2011-2012 based on engineering analyses design [1]. The near to final design of the diagnostic rack is checked for the consistency to electromagnetic, thermal, seismic and fatigue loads in the paper. To work in the specific ITER environment imposes a restricted list of materials demanding a more careful design of optical elements to accommodate thermal expansion. A special attention is paid to the deformed shape of the mirrors under operating loading conditions and its effect on optical system performance that is an actual question for all optical systems with mirrors designed for the ITER.

[1] E.E.Mukhin, et al., “The ITER divertor Thomson scattering system: engineering and advanced hardware solutions” JINST 7 C02063 (2012)

P3-044 **Ceramic breeder for fusion nuclear reactors: thermo-mechanical tests on pebble beds**

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Mechanical and thermal properties of the lithium ceramic pebble beds, such as the lithium orthosilicate or lithium metatitanate are key issues to be investigated in the framework of fusion power technology, for the reason that they are possible candidates in the design of breeder blankets. The aim of paper is to evaluate experimentally the thermal conductivity of ceramic pebble beds versus the temperature and the compressive strain. To the purpose a thermo-mechanical characterization of pebble beds made of a large available material (alumina), having different diameters, was done by means of an experimental test campaign considering a wide range of temperatures and compression forces. The experimental tests were performed at the DICI-University of Pisa, based on the principle of the steady state axial heat transfer through different materials (one of known conductivity) method. Moreover several tests on the ceramic breeder have been performed. The aim of the tests is to measuring the effective thermal conductivity of the pebble beds in the conditions foreseen for the breeding blanket. The obtained results may contribute to the creation of a database of the thermal conductivity of pebble beds which can be used for the design and analysis of fusion breeder blankets.

P3-045 **Effect of Electromagnetic Coupling on MHD Flow in the Manifold of Fusion Liquid Metal Blanket**

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In fusion liquid metal (LM) blanket, magnetohydrodynamics (MHD) effects will dominate flow patterns and heat transfer characteristics of the liquid metal flow. Manifold is a key component in LM blanket in charge of distributing or collecting the liquid metal coolant. In this region, the complex three dimensional MHD phenomena will be occurred, and the velocity pattern, pressure and flow rate distribution may be dramatically influenced. One important aspect is the electromagnetic coupling effect resulting from an exchange of electric currents between two neighboring fluid domains that can lead to modifications of flow distribution and pressure drop compared to that in separated channels. Understanding the electromagnetic coupling effect in manifold is necessary in making an optimize design for liquid metal blanket. In this work, a numerical study was carried out by using a code named MTC 2.0 to investigate the effect of electromagnetic coupling on MHD flow in a manifold region. The typical manifold geometry in LM blanket was considered, a rectangular supply duct entering a rectangular expansion with toroidal field oriented along the expansion direction, finally feeding into 3 or 4 rectangular parallel channels stacked in the field direction. The influence of electromagnetic coupling effect on the MHD flow partitioning was analyzed, and the 3D pressure drop was evaluated.

P3-046 **Progress on a coupled Systems Code-CFD MHD solver for fusion blanket design**

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Fusion blankets are required to operate in a harsh environment under the influence of a number of interdependent and synergistic physical phenomena, working across several length scales. For magnetic confinement reactor blanket designs using a conducting fluid as coolant/breeder, the difficulties in flow modelling are particularly severe due to interactions with the large magnetic field.

Blanket analysis is an ideal candidate for the application of a code coupling methodology, with a thermal hydraulic systems code modelling portions of the blanket amenable to 1D analysis, and CFD providing detail where necessary. A systems code, MHD-SYS, has been developed and validated against detailed analysis of the LLCB blanket concept. The code shows good agreement in the prediction of MHD pressure loss and the temperature profile in the fluid and wall regions of the blanket breeding zone. A suitable CFD MHD solver is described and a methodology for coupling the two codes via TCP socket connections detailed.

Keywords:
Fusion blankets, MHD, Systems codes, CFD, OpenFOAM, Coupled codes

P3-047 **Design and Setup of a Testing Device to investigate a reduced sized attachment system mock up for the ITER EU HCPB-TBM under different mechanical loading conditions**

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The EU-Helium Cooled Pebble Bed Test Blanket Module (HCPB-TBM), which will be located inside an equatorial port plug of ITER, is connected to the shield by an attachment system. The recent design approach followed in KIT of the attachment system consists of a hollow cylinder located at the center of the back plate. One of the most demanding loading conditions for the TBM will be high electromagnetic forces acting on the TBM box during operation. In order to measure these forces and therefore support the design of future TBMs, a force measuring system for the attachment system is developed. In order to check the capability of the measuring system, a testing device has been designed and built. The design of the testing device and corresponding mock ups is based on FEM analyses to ensure that relevant conditions can be generated on reduced sized mock ups. The mock ups are equipped with a strain measurement system to record strains at several points on the cylinder. The testing device has a modular setup and is therefore able to accommodate different mock ups. In a first step, a simple tube is used in order to calibrate and test the setup. In a second step, a mock up representing a TBM box with attachment system will be installed.

This paper presents the design and setup of the testing device and describes the different mechanical and electrical components as well as the measurement and data acquisition system. In addition, the design and fabrication of the two first mock ups as well as their relevance with regard to the cylindrical attachment system are discussed.

P3-048 **Construction of PREMUX and preliminary experimental results, as preparation for the Helium Cooled Pebble Bed Breeder Unit mock-up testing**

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One of the European blanket designs for ITER is the Helium Cooled Pebble Bed (HCPB) blanket. The core of the HCPB-TBM consists of so-called breeder units (BU), which encloses beryllium as neutron multiplier and lithium orthosilicate (Li4SiO4) as tritium breeder in form of pebble beds. After the design phase of the HCPB-BU, a non-nuclear thermal and thermo-mechanical qualification program for this device is running at the Karlsruhe Institute of Technology.

Before the complex full scale BU testing, a pre-test mock-up experiment (PREMUX) has been constructed, which consists of a slice of the BU containing the Li4SiO4 pebble bed. PREMUX is going to be operated under highly ITER-relevant conditions and has the following goals: (1) as a testing rig of new heater concept based on a matrix of wire heaters, (2) as benchmark for the existing Finite Element Method (FEM) codes used for the thermo-mechanical assessment of the Li4SiO4 pebble bed, and (3) in-situ measurement of thermal conductivity of the Li4SiO4 pebble bed during the tests. This paper describes the construction of PREMUX, its assembly steps and the experimental campaign planned with the device. Preliminary results of the test runs performed up to date are reported and compared with first analyses completed with the FEM codes.

P3-049 **Evaluation of hydrogen isotope absorption/diffusion coefficient of CVD-SiC in high temperature**

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SiC monolithic and SiC/SiC composite are candidate materials of blanket in fusion reactors as silicon car bide are low activation, high heat resistance, and low hydrogen permeability. We have been working on measurements of permeability of the hydrogen isotopes through SiC materials. First series of experiments gave complicated results, which were much higher than previous works. Following it, we conducted the measurement of solubility and diffusion of deuterium in SiC powder and fiber that consist SiC/SiC composite to understand the complicated behavior of hydrogen isotopes in it. In this paper, deuterium diffusion coefficient evaluation by solubility and diffusivity experiment for CVD-SiC will be presented. The result shows that diffusion coefficient is at least 2-order smaller than those obtained by permeation experiments. Thus it is confirmed that permeation through the CVD-SiC is also mainly depend on the permeation speed through grain boundary. It is also found that measurement of grain size and its distribution is important as assumption of grain size largely affects evaluation of diffusion coefficient in solubility and diffusivity experiment.

P3-050 **Tritium management issues and anti-permeation strategies for different DEMO breeder blanket options**

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In a DEMO like machine, the accepted tritium losses from the breeder blanket (BB) through the steam generator (SG) shall be as low as 2 mg/d, which are more than 5 orders of magnitude lower than production rate of about 360 g/d. This imposes a very effective mitigation strategy, taking into account and balancing the size of the tritium processes in the breeder and cooling loops, and the efficiency of anti-permeation barriers.

A numerical study has been performed using the FUS-TPC tritium permeation code that computes all tritium flows and inventories considering the design and operation of the BB, the SG, and the tritium systems. Slightly different FUS-TPC variants have been developed to reflect the specificities for tritium migration in three different blanket candidates (HCPB, HCLL, WCLL). Such numerical tool allows studying the tritium permeation issues and evaluating mitigation strategies.

Many scenarios were computed for all the three blankets concepts, varying the tritium systems throughput and efficiency, and also studying the influence of the permeation regimes through the BB and SG assumed to be either diffusion-limited (with varying the permeation reduction factors) or surface-limited. For all blankets, a very efficient way to limit tritium releases would be to ensure a surface-limited permeation regime for both the BB and SG. Under these conditions, tritium losses could be kept below the 2 mg/d target while keeping the tritium processes requirements reasonable. Under diffusion limited regime, even if very efficient and large tritium extractions systems are implemented, significant permeation reduction factors will still be required to maintain the size of the tritium processes feasible. For each blanket candidate, workable scenarios will be reported and analysed with regards to the technological requirements. Specific anti-permeation strategies will be discussed for the different DEMO blankets to ensure a safe and efficient tritium management.

P3-051 **Experimental study of instabilities in a quasi-2D MHD duct flow with near-wall jets**

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In liquid metal (LM) fusion blanket duct flows subjected to a strong transverse magnetic field, quasi-two-dimensional (Q2D) velocity profiles with near-wall jets often develop, containing multiple inflection points where instabilities tend to form. Inflectional instabilities that form in the bulk flow can interact with one another or with the side layer to form more complex secondary instabilities, and the formation and evolution of these instabilities can cause a duct flow to transition from laminar flow to full or Q2D turbulence. Since these flow regimes have very different heat/mass transfer and pressure drop characteristics, there is much interest in learning what conditions cause the different types of instabilities and flow regime transitions. Therefore, we have constructed an experiment that electromagnetically induces a Q2D velocity profile with near-wall jets in an insulated closed rectangular duct, which is filled with LM and bathed in a strong transverse applied magnetic field. The production of Q2D wall jets through the application of a steady current across a portion of the Hartmann walls (surfaces perpendicular to the applied magnetic field) causes two shear layers to form at the current injection electrodes (inflection point locations), one at the anode and one at the cathode. The velocity field on the Hartmann walls is measured and recorded with a 2D array of very small (0.25-mm diameter) electric potential probes embedded in the walls,which produce no significant hydrodynamic or electromagnetic disturbance in the flow.The probe array allows for the detailed analysis of the fluid motion in the vicinity of the electrodes, near the walls, and in the bulk as the applied current and magnetic field are altered. In this paper we present first experimental observations and analysis of inflectional instability development and the formation of Q2D eddies in a LM magnetohydrodynamic duct flow with induced near-wall jets using this velocimetry technique.

P3-052 **Characterizing Pressure Equalization in MHD Flow in a Rectangular Duct with an Insulating Flow Channel Insert**

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The Dual-Coolant Lead-Lithium (DCLL) concept incorporates a Flow Channel Insert (FCI) to mitigate MHD pressure drop and to serve as thermal insulator. To reduce mechanical stress in the FCI, pressure equalization openings, such as holes or a slot, in the FCI wall are designed to equalize pressure in the flowing liquid metal (LM) between the bulk flow inside the FCI box and that in the thin gap between the FCI and the outer ferritic steel duct. In this study, the MHD flow and associated pressure equalization effects are simulated with a recently developed powerful numerical code (HIMAG) in 3D. Two pressure equalization mechanisms have been identified and studied: one is due to LM flow through the openings and the other is due to induced electric currents flowing across the non-ideally insulating FCI wall. The second effect appears to provide a more effective way of pressure equalization compared to the purely hydrodynamic mechanism. Parametric studies have been performed to address the impact of the FCI electrical conductivity and slot size. Finally, recommendations on the FCI design are proposed to achieve more effective pressure equalization.

P3-053 **Design and Construction of a Multipurpose Laboratory Scale Apparatus to Investigate Hydrogen Isotopes Behaviour In Pb15.7li and Permeation Technology**

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Lead lithium eutectic alloy, thanks to its unique properties, being tritium breeder (Li) combined with neutron multiplier (Pb) in the same material, represents one of the candidate breeder materials for fusion reactors breeding blankets. The knowledge of the tritium transport parameters in lead lithium is fundamental for the design of the HCLL (Helium Cooled Lead Lithium) blanket. More in detail the inventory of tritium in fusion reactors blankets and the permeation of tritium into the blanket coolant or in the environment is a function of its solubility and diffusivity in the lead alloy Pb15.7Li.

Several experiments, devoted to investigate the function which links the tritium solubilised in lead lithium with the corresponding tritium partial pressure at equilibrium were carried out in the past, but considerable uncertainty still remains. Taking into account the big impact of such issues on the lead lithium eutectic based blanket performance, an experimental activity devoted to achieve reliable and agreed data was launched by F4E, and a multipurpose laboratory scale facility was designed and constructed in ENEA Brasimone. The facility will be capable not only to perform hydrogen transport parameters measurements in lead lithium but also to qualify hydrogen sensors and permeation barriers in lead lithium. The facility is here presented together with the first qualification activities carried out.

P3-054 **Burnup analysis and fissile fuel breeding with a Uranium-Plutonium cycle in the molten salt blanket in a FFHR**

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Progress on the fusion-fission hybrid reactor (FFHR) brings fusion a viable energy source in foreseeable future. Energy multiplication in a FFHR makes a much easier prerequisite for the fusion reaction than a fusion reactor. In the previous work, a hybrid reactor with a molten salt blanket was designed. The molten salt reactor has advantages on heat transfer and post-processing of the spent fuels. The molten salt is consisted of F-Li-Be, with nuclear fuels dissolved in it. Uranium-plutonium fuels were chosen for study. The uranium-plutonium fuels were added into a F-Li-Be molten salt zone of 40cm thickness and by changing the component of each nuclear element, the appropriate blanket energy multiplication factor and TBR can be obtained. In the current work, the burnup characteristics of the molten salt blanket in a FFHR was studied. A whole burnup life of 10 years has been analyzed. The burnup analysis of the molten salt blanket was carried out by the COUPLE2 code. Through the burnup analysis, the breeding of the fissile fuel 239Pu and the transmutation of the minor actinides were also studied.

P3-055 **Development of the breeding blanket and shield model for the fusion power reactors system SYCOMORE**

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SYCOMORE, a fusion reactor system code based on a modular approach is under development at CEA. In this framework, this paper describes the models which have been developed and relevant sub-modules which have been implemented to model the main aspects of the breeding blanket and shield block of the system code (tritium breeding ratio, peak energy deposition in toroidal field coils, reactor layout and power flow, blanket pressure drops and materials inventory,…). This module is constituted by three sub-modules: the "blanket layout" sub-module, the neutronic sub-module, the thermal hydraulic and thermo-mechanic module. The power flow module has also been developed which is directly linked to the blanket thermo-dynamic performances. For the "blanket layout" and thermal thermo-hydraulic sub-modules explicit analytic models have been developed and implemented; for the neutronic sub-module, on the other hand, appropriately surrogate models based on neural networks have been built. Presently, relevant models for the Helium Cooled Lithium Lead and for the Water Cooled Lithium Lead blanket have so far been implemented. Sub-modules have been built in a way that they can run separately or coupled into the breeding blanket and shield module in order to be integrated in SYCOMORE. In the paper the input/output parameters of each sub-module are reported and relevant models discussed. The application to previous studied reactor models (PPCS model AB, DEMO-HCLL 2007) is also presented.

P3-056 **Parametrical analysis of HCLL and HCPB TBM tritium transfer model with TMAP7**

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Tritium availability in fusion reactors is one of the major issues to be solved. Because tritium is not a resource available from natural sources and its emission from the facility should be minimized, to keep the material balance in the entire plant and to control its inventory and safety is mandatory. Tritium can be produced within the reactor by nuclear reactions with lithium contained in a blanket. The European concepts of Test Blanket Module to be tested in ITER are Helium Cooled Lithium Lead (HCLL) and Helium Cooled Pebble Bed (HCPB). A tritium transfer model for this two concepts was developed, which includes the TBM and the extraction loop. An analysis on tritium permeation, extraction and inventory in time were done by using the TMAP7 code (Tritium Migration Analysis Program). The parametric study includes different irradiation scenarios and physical configurations, varying size, extraction efficiency, thickness of components and structural material.

P3-057 **Development of an HCLL TBM Configuration And Ancillary Systems Dynamic Transfer Model with ECOSIMPRO®**

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The dynamic and global inventory of tritium must be strictly controlled in the ITER.This requirement applies also to the Test Blanket Modules (TBM) and related auxiliary systems. This control will affect the operational availability, the operating modes of the facility and the design of some of its units and systems. Accomplishing this complex task as required by ITER is one of the motivations behind this paper. To do this, the development of a simulation model to determine the Tritium transfer phenomena and the tritium inventory in the different units is fundamental. One of the main goals of this simulation model is to determine the amount of tritium per unit time that can be collected, accounted for and routed out of the TBS auxiliary systems to the tritium systems of the ITER Tritium Plant.

ECOSIMPRO®, born in the frame of ESA aerospace program and QA certified for use in a range of fields, is developed by the Spanish company Empresarios Agrupados. It has been preliminarily selected as the simulation software for developing thedynamic model of the ITER HCLL-TBM and ancillary systems. The mathematical models implemented in the simulation code are able to determine the different tritium flows through the TBM and the ancillary systems based on the basic principles that govern the diffusion, solubilisation and dissociation/recombination phenomena

The following simulation models have been developed:

-Helium Cooled Lithium Lead (HCLL)TBM

-LiPb loop and Tritium Extraction Unit (TEU)

-Tritium Removal System (TRS)

-Helium Coolant System (HCS)

-Coolant Purification System (CPS)

The study is developed through a parametric analysisofdifferent operating conditions of the TBM ancillary systems produced byvarious ITER irradiation scenarios and different TBS sizes and materials. The main results will be presented and discussed with some detail in this paper.

P3-058 **Modelling a supercritical CO2 power cycle for nuclear fusion reactors using RELAP5-3D®**

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A supercritical recompression CO2 power cycle has been simulated using the system code RELAP5-3D®. This code is being developed by INL and has traditionally been used in the simulation of operational and accidental transients in fission nuclear plants. The aim of the work presented here, developed within the framework of the Spanish Fusion Technology Program Consolider TECNO_FUS, is to take advantage of the simulation capabilities of RELAP5-3D® in a field where little if any experience exists in the use of the code; i.e., the simulation of the heat fluxes and the thermodynamic cycle that, in a fusion power plant, will convert thermal power from plasma into mechanical power as a previous step to electricity generation. Code capabilities that make it suitable for this purpose are, for instance, the compressor model and the libraries of fluid properties (among them CO2 and LiPb).

The reference plant for the simulation is the one being designed under TECNO_FUS, which is the Spanish proposal for DEMO. The model of the plant includes the primary coolant systems, i.e. helium and LiPb in the Spanish dual coolant modular design (doble refrigerante modular, DRM), compressors, turbine and heat exchangers (Prited Circuit type).

After the model has been set-up, several steady-state calculations have been run to test the performance of the model. After designing some minimal control features and adjusting their parameters, a few transient calculations have been run in order to demonstrate the capabilities of code and model. Finally, strengths and weaknesses of code and model are highlighted, along with some conclusions on their suitability for fusion technology applications.

P3-059 **Hydrogen solubility in liquid lithium-sodium alloy**

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Liquid alloy ($Li_{0.9}Na_{0.1}$ to $Li_{0.2}Na_{0.8}$) was synthesized in a 1mm thick iron capsule from high purity metal ingots (>99.9%). The capsule was sealed by a metal cap and inserted in an annular furnace. Low pressure hydrogen gas was introduced in the furnace to feed hydrogen into the alloy by permeation through the iron wall. Afterward, the capsule was heated (673-873 K) in high vacuum to measure hydrogen permeation leakage through the wall with a mass analyser. Changing hydrogen concentration and temperature, hydrogen solubility (Sieverts' constant) of the alloy was estimated. Hydrogen solubility decreased depending on the sodium concentration in the alloy; for example, hydrogen solubility of Li0.9-Na0.1, Li0.5-Na0.5 and Li0.2-Na0.8 alloys were almost 1/2, 1/5 and 1/30 of that of pure lithium at 873 K respectively. The decrease in hydrogen solubility – equivalent to the increase in equilibrium hydrogen pressure- may make tritium recovery by permeation method functional.

P3-061 **Analysis of the thermo-mechanical behaviour of the DEMO Water-Cooled Lithium Lead breeding blanket module**

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Within the framework of the European DEMO Breeder Blanket Programme, a research campaign has been launched by the University of Palermo, ENEA-Brasimone and Commissariat à l'Energie Atomique with the specific aim to theoretically investigate the thermo-mechanical behaviour of the Water-Cooled Lithium Lead (WCLL) breeding blanket module. Attention has been focussed on the water-cooled DEMO1 blanket vertical segment, in order to investigate its thermo-mechanical performances under the Normal operation and Over-pressurization loading scenarios.

The research campaign has been carried out following a theoretical-computational approach based on the Finite Element Method (FEM) and adopting a qualified commercial FEM code. Two realistic 3D FEM models of the WCLL blanket module have been developed, the first representing its central poloidal-radial region, including one breeder cell in the toroidal direction and all the five cells in the radial one, the second simulating its entire segment box, without the breeder and its relevant cooling tubes.

Two set of uncoupled steady state thermo-mechanical analysis have been carried out with reference to the investigate loading scenarios, adopting the first model for the Normal operation scenario and the second for the Over-pressurization one. Proper thermal and mechanical loads and boundary conditions, have been assumed. In particular, it has been assumed that under the Normal operation scenario (level A) the module undergoes both 15.5 MPa coolant pressure on its cooling channels walls and thermal deformations due to the flat-top plasma operational state thermal field, while under the Over-pressurization scenario (level D), induced by a coolant leak, the module undergoes 15.5 MPa coolant pressure on its internal walls while it operates at the normal operation thermal level. Results obtained are presented and critically discussed according to the SDC IC and RCC MRx codes.

Topic C Fuel Cycle

P3-062 **Alternative Analysis for Fuel Storage and Delivery in the ITER Tritium Plant**

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The Tritium Plant is one of the important fundamental facilities for the fuel cycle not only in ITER but also in a future fusion power plant. The Tritium Plant plays roles in delivering fuel gases, such as tritium and deuterium gases, and in handling all tritium-contaminated gases to remove impurities from hydrogen isotopes, to separate each species among the hydrogen isotopes, to store the high concentrated tritium and tritium-contaminated deuterium gases, and to prevent the release of all radioactive gases to the environment. The Storage and Delivery System (SDS) of the ITER Tritium Plant has to safely handle the fuel gases, mainly tritium-contaminated gases. One of the main objectives of the SDS is to deliver the main fuels, such as highly concentrated tritium and deuterium gases to the Fuelling System directly delivering fuels for gas puffing, pellet injection, and neutral beam injection. In this study an alternative analysis for fuel storage and delivery process will be performed to satisfy the required fuelling species and rates under the inventory limitation of the tritium-contaminated gases in a safety-manner. Three different kinds of time-scales will be considered as two-year, two-week, and daily intervals for the ITER plasma operations. Various process concepts will be proposed based on pathways of flows of the fuel gases and configurations of equipments. Each process concept will be analyzed with respect to required numbers of main equipments, such as tritium storage beds, buffer vessels, and fuel delivery pumps. There is trade-off relationship between the required number of tritium storage beds and the required number of buffer vessels under value engineering. One of the concepts for the fuel storage and delivery process will be selected and how to operate in detail will be determined.

P3-063 **Tritium Transport Modelling of Libretto-4 & 5 Transients**

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The last series of LIquid BReeder Experiment with Tritium Transport Option (LIBRETTO-4 and 5) were performed to test in-pile tritium breeding/release from liquid lead-lithium eutectic and permeation through a pre-oxidized ferritic-martensitic steel EUROFER97 at the HFR of Petten. Two identical experimental capsules LIBRETTO-4/1 and 4/2 were irradiated at 300ºC and 500ºC respectively over 700 Full Power Days (FPD) providing in-pile continuous data. The tritium breeding alloy filled the space between two tubes. The first tube introduced two gas-lines with a mixture of He with 0.1% of H2 (and nominal 2.8 bar and 100ml/min). One to sweep the LiPb plenum and the other one to purge the gap between the EUROFER wall of the second tube and a third containment. The LIBRETTO-5 had a bigger volume of lead-lithium and was irradiated over 500 FPD at the temperature range of (300-500ºC). Different transients were performed modulating temperature and hydrogen concentrations.

The physical phenomena determining tritium released and permeated quantities under radiation conditions are analyzed at different transients through benchmarking of tritium residence times and implemented models with experimental values. Furthermore, the reliability of TMAP7, ECOSIMPro® and COMSOL Multiphysics© as TT modelling tools are being proved by this verification process.

P3-064 **Hydrogen extraction characteristics of high-temperature proton conductor ceramics for hydrogen isotopes purification and recovery**

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To develop high-efficiency and economical hydrogen isotope purification and recovery processing for Tritium Extraction System of TBM, hydrogen extraction characteristics of a electrochemical hydrogen pump using perovskite-type CaZr0.9In0.1O3 α as high-temperature proton conductor were studied, because one of its driving forces of hydrogen transportation is electric potential difference and can extract hydrogen isotopes selectively from a blanket sweep gas with low-pressure hydrogen isotope gaseous.

The extraction of hydrogen from dilute hydrogen, water vapor electrolysis and decomposition of methane were demonstrated for fusion engineering applications. One-end closed CaZr0.9In0.1O3 α tube with the size of 20mm in outer diameter, 17mm in inner diameter and 200mm in length was selected because of its high chemical stability, mechanical strength, and durability in spite of lowest conductivity in a series of perovskite-type proton conductors. The hydrogen pumping characteristics were evaluated over the temperature range from 873K to 1073K by helium with 10-5000ppm hydrogen, finding a maximum hydrogen evolution rate. The hydrogen recovery efficiency was more than 99% in the mixture of He with hydrogen over 1000 ppm at temperature over 973K and applied voltage over 2.5 V DC. The extraction of hydrogen could be operated with a current efficiency close to unity, while some electronic conduction appeared in the water vapor electrolysis.

P3-065 **Towards a physics-integrated view of the fusion fuel cycle**

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The fusion fuel cycle is a central element of a DT fusion machine. It comprises the fuel injection and gas based plasma control systems, the torus exhaust vacuum pumping systems and the tritium plant as well as the tritium breeding systems. All these are technical systems which are designed against requirements given by the fusion machine operation, and, finally, by plasma physics conditions. However, the design of the fuel cycle sub-systems in the past has been developed separately, only relying on a small number of engineering interface parameters, of which the steady-state fuelling rate being the most important one. The numerical quantities used for the interface parameters are generally of high impact on the final design, so that any major change of these will have to be followed by a rigorous re-design.

In recent years, the awareness of this complex interaction has grown and considerable progress has been made to better interlink physics and technology issues in the interfaces of the fuel cycle with the plasma, viz the pellet injection systems and the divertor and its gas exhaust vacuum pumping system. The paper will start with a short introduction of the systems and outline how their design has evolved following the progress in physics understanding. It will present recent achievements in divertor gas flow modeling and in improved understanding of the physics of core fuelling; these results have been elaborated within the inner fuel cycle modelling project under the EFDA ITER Physics Programme. Finally, the limits of the current system designs are described.

P3-066 **Research activities related to water detritiation at ICIT Rm. Valcea**

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The Tritium Removal Facility is an experimental pilot plant for deuterium and tritium separation. This installation is a national interest unit, having as main goal the development of the heavy water detritiation technology and also to monitor the behaviour of various materials and equipment working with tritium and at cryogenic temperatures. There are also a number of support activities and a tritium research program which are presented as an overview of the ICIT new research tasks.

This work is focused on the presentation of ICIT research activities, perspectives and its capability related to water detritiation technologies and also to the development of research on issues in the field of nuclear fusion.

P3-067 **Tritium Removal Facility for processing of a large range tritiated water waste**

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Tritium Removal Facility was developed by ICIT with the aim to promote a technology for detritiation of heavy water. Another purpose of the facility is to recover the tritium from various tritiated water waste. The installation was built at semi-industrial scale, and the technological process is based on LPCE-CD separation followed by tritium storage on titanium or uranium bed. To achieve the tritium recovery from water waste, it has been designed a module based on CECE separation process. In Romania the research results will be applied to Cernavoda Nuclear Power Plant (two CANDU reactors operation). The results could also be useful for international fusion research program. The paper presents a brief description of the TRF developed by ICIT and a set of simulation data for isotopic exchange process, expected to be realized in this installation.

P3-068 **High gain and frequency ultra-stable integrators for long pulse applications**

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Inductive pickup loops are one of the primary magnetic diagnostics used in modern fusion and pulsed power concepts. To convert the direct voltage measurements from the inductive pickup loop to a measurement of magnetic field or current, the loop voltage must be integrated. Several factors make the integration difficult, especially for long-pulse applications requiring integrator stability for operational timescales from seconds to hours.

Eagle Harbor Technologies (EHT) has recently developed a new Ultra-Stable integrator that has a wide range of applications within the plasma science and pulsed power communities. It is the only integrator that meets (and dramatically exceeds) the ITER stability specification. With a sub 100 ns rise time, the integrator can integrate pulses ranging from 100 ns to 100 hours. Because of its extremely high gain and high frequency of operation, the integrator has an unprecedented effective dynamic range. For example, if given a 1 second square pulse input, using a 12 bit digitizer, the integrator can produce a post processed signal with 100 ns temporal resolution, and 22 bits of dynamic range information. This capability allows for extremely small high frequency detail to be resolved on much larger and slower signals.

The EHT Ultra-Stable integrator measures with high precision both extremely small fields, and extremely large fields. For example, if connected to a loop with a 0.1 m^2 area (100 turns 3 cm in diameter) wound on ferrite with a mu of 1000, the integrator would produce a 1 mV output when measuring a 100 pT field. When measuring medium to large fields, small air core pickup coils would be used. Since air core coils do not suffer saturation issues, the EHT integrator can measure very large currents, for very long times, without the usual droop and saturation issues encountered when using standard current monitors/transformers. There is no inherent limit to the maximum field size or current that can be measured. The new EHT integrator system, along with characterization data, will be presented.

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P3-072 **Evaluation of tritium transport in nuclear fusion materials under irradiation at LIBRETTO-4**

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For future nuclear fusion reactors a Tritium production and extraction Plant is a need in order to guarantee tritium self-sufficiency. Transport phenomena through the Test Blanket Modules (TBM) are complex and involve many physical properties and parameters that need to be understood and accurately assessed for the design of an economic and safety Tritium fuel cycle. There is no practical external source of tritium, therefore in magnetic nuclear fusion as inertial nuclear fusion, achieving a proper Tritium Breeding Ratio (TBR) is a key goal. LIBRETTO (Liquid Breeder Experiment with Tritium Transport Option) experiments were designed and accomplished in order to study i) permeation barriers efficiency and ii) tritium transport, under irradiation in a high flux reactor (HFR). For LIBRETTO-4 the two experiments (-4/1, -4/2) were almost identical but operated at two different temperatures (300-350, 500-550 ºC) providing in-pile continuous and comparable data. This report focuses on the numeric modelling approaches to reproduce tritium release-rate data, both permeation and desorption at LIBRETTO-4 capsule 2 at 300-350 ºC. Tritium transport parameters derived during this work are reported and discussed.

Acknowledgments: The main author thanks L. Magielsen and A. Fedorov from NRG (Nuclear Research and consultancy Group, Petten, The Netherlands) for providing the experimental data of LIBRETTO-4 used in this work.

P3-073 **Modelling of ITER divertor pumping system during various operational scenarios via kinetic theory**

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The most recent reference design of the ITER torus exhaust pumping system [1] is based on 6 cryopumps, directly connected via ducts to the torus. The six large cryo-sorption pumps, working in several pumping and regeneration sequences evacuate the neutralized gas from the divertor to balance the fueling and remove the fusion helium and impurity exhaust. In all cases the gas passages through the divertor cassettes and torus pumping ducts form a complex, labyrinth type, network of conductances in different pressure regimes. To monitor and optimize the overall operation it is important to model this network. Computational algorithms for solving the gas flow through complex distribution systems are well developed only in the viscous (or hydrodynamic) regime. Corresponding algorithms for solving gas distribution systems under any degree of gas rarefaction are very limited and the few available ones are based mainly on empirical expressions. Recently, a more sophisticated algorithm based on kinetic theory principals has been developed and successfully implemented to simulate in the whole range of the Knudsen number gas distribution systems consisting, however, only of long and moderately long channels [2,3]. In the present work, this recently developed algorithm is further developed and advanced to model and simulate the new configuration of the ITER divertor exhaust pumping system. The analysis is based on an adequately dense kinetic data base providing the conductance of various piping elements in terms of pressure ratio, length and rarefaction parameter. This data base is successfully integrated in the network algorithm. Since a kinetic approach is implemented, the analysis is valid and the results are accurate in the whole range of the Knudsen number, while the involved computational effort remains small taking into consideration that the data base has already been developed. The presented modeling refers to two operational scenarios of the reactor, related to the burn and dwell phases. A complete solution of the rarefied gas flow through the pipe network is provided by tabulating the resulting throughputs and conductances through the each pipe element as well as the pressure heads at each junction of the network. The proposed algorithm may be used to simulate other ITER operation modes as well.

Topic D Materials

P3-074 **Evaluation of applicability of a laser-based distance meter to measurement of Li jet thickness for the IFMIF/EVEDA project**

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In the Engineering Validation and Engineering Design Activities (EVEDA) of the International Fusion Materials Irradiation Facility (IFMIF), a device to measure thickness variation of the Li target with precision of 0.1 mm must be delivered. For this purpose, we newly focused on a laser-based distance meter. This paper describes the result of an applicability test conducted in the Osaka University Li Loop. In the experiment, thickness variation of a Li jet (10 mm in thickness and 70 mm in width) was measured in 50 to 180 mm downstream region from the nozzle in the velocity range of 10 to 15 m/s at the operation temperature of 573 K. The laser was emitted to the Li jet through the view glass. To measure specular Li surface, an incident laser must be returned to the sensor. Such a condition is satisfied when a normal vector of the surface faces in the direction to the sensor. It is considered this condition is achieved at wave crests or bottoms. Even while we cannot continuously obtain the Li level data even with high sampling frequency such as 500 kHz, we can analyze the Li level statistically. To evaluate the applicability of the device, the measurement precision of the Li level was evaluated. It is considered the precision can be evaluated by thickness displacement in the sampling period of 2μs. Assuming wave shape is sine wave, the displacement in that period is less than 1 μm (almost flat) in the velocity range of 10 to 15 m/s. Therefore, the precision can be defined by the standard deviation of the histograms of the thickness displacements with the period of 2 μs. As a result, the precision was approximately 9 μm. Thus, we concluded that the laser-based distance meter is applicable to measurement of the Li target thickness.

P3-075 **Mechanical Properties of Similar and Dissimilar Weldments of RAFM Steel and AISI 316L(N) Stainless Steel Prepared by Electron Beam Welding Process**

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Reduced Activation Ferritic Martensitic (RAFM) Steel is a candidate structural material for the fusion power reactors. The Indian RAFM (IN-RAFM) steel has been chosen for fabricating Indian Test Blanket Module (TBM) to be tested in ITER. Chemical composition of the INRAFM steel is similar to that of modified 9Cr-1Mo steel except Mo and Nb which were replaced with W and Ta in order to reduce the induced activity. Further, active elements like Cu, Mn, P, B, S are reduced in this steel. Volume of the weld metal in actual component is to be reduced and considering this aspect, an advanced welding process is chosen for development of welding procedure for similar and dissimilar materials. Weld joints of similar and dissimilar materials of INRAFM steel and AISI 316L(N) SS were made using Electron Beam Welding (EBW) process. Weld joints were radiographed and found to be defect-free. Hardness of the INRAFM steel weld metal was higher compared to its base metal and width of the heat affected zone (HAZ) was found to be ~ 0.5 mm. Tensile strength of the cross weldment specimen is 684 MPa with location of fracture being base metal; tensile strength is comparable to that of the base metal. Charpy V-notch impact tests of the weld zone were conducted using subsize specimens and the values are 96 and 110 J in as-welded condition and after post weld heat treatment (PWHT) at 750ºC/90min respectively. These values are comparable to that obtained for base metal. For the dissimilar weld joint between INRAFM steel and 316LN stainless steel, hardness value is found to be similar to that found for the INRAFM steel weld metal in as-welded condition and the value is ~450-480 VHN. Tensile strength of similar INRAFM steel weld joints and dissimilar weld joints is found to be comparable but higher than the similar SS weld joint. Microstructure and mechanical properties of similar and dissimilar weld joints prepared by EBW process were characterized and results will be discussed in this paper.

P3-076 **Non-linear Failure Analysis of HCPB Blanket for DEMO Taking into Account High Dose Irradiation**

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For the European helium cooled pebble bed (HCPB) blanket of DEMO the reduced activation ferritic martensitic steel EUROFER has been selected as structural material. During operation the HCPB blanket will be subjected to complex thermo-mechanical loadings and high irradiation dose. Taking into account the material and structural behavior under these conditions is a precondition for a reliable blanket design. In recent non-linear structural analyses the design of the HCPB Test Blanket Module (HCPB-TBM) has been considered [1]. As the irradiation dose the HCPB-TBM will be subjected to in ITER is relatively low, the irradiation effects on the material behavior of EUROFER were out of consideration. In DEMO the irradiation dose will be much higher and irradiation effects cannot be excluded. For considering high dose irradiation in structural analysis of DEMO blanket, the coupled deformation damage model extended recently taking into account the influence of high dose irradiation on the material behavior of EUROFER [2] has been implemented in the finite element code ABAQUS. Using this implementation non-linear finite element (FE) simulations of DEMO HCPB blanket have been performed considering the HCPB-TBM design of previous analyses as a reference. While the thermal and mechanical boundary and loading conditions remain the same as in previous analyses, the loading scenarios are adjusted considering pulsed and steady state DEMO operation. The irradiation dose rate required at each position in the structure as an additional loading parameter is determined in preceding neutronic analysis using the MCNP code and proper DEMO boundary conditions. The results of the FE simulations are evaluated and discussed considering ratcheting and damage at most critical highly loaded areas of the structure as well as the impact of high dose irradiation on these failure modes.

References

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P3-077 **Study on Li diffusivity in lithium metatitanate with excess lithium (Li2+2xTiO3+y) at high temperature**

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Lithium metatatitanate with excess lithium Li2+2xTiO3+x (Li/Ti > 2.0) has been developed as an advanced ceramic breeder material for a blanket module of a future DEMO power reactor because of its higher Li density and stability in a reducing atmosphere than stoichiometric b-Li2TiO3. b-Li2TiO3 phase (Li2SnO3 type rock salt structure, space group 2C/c, Z=8) has been reported to be non-stoichiometry in the range between 47 and 51 mol% TiO2, in Li2O-TiO2 phase diagram. The past study by Hoshino (Fusion Eng. Des. 82 (2007) 2269–2273) reported that Li2+xTiO3 has the higher sum of the partial pressures of the Li containing species than Li2TiO3. It is necessary to understand the vaporization behavior at high temperature because the change of Li density due to the vaporization may have major impact on tritium breeding ratio (TBR). And the vaporization behavior is closely related to the diffusion process of Li atom. Therefore, the diffusivity of Li2+2xTiO3+x at high temperature was studied in this study. At first, nonstoichiometric lithium metatitanate with various Li/Ti ratios from 1.8 to 2.3 at mixing were synthesized by sol-gel process. High temperature neutron diffraction was carried out by high efficiency high resolution measurements (HERMES) from room temperature up to 1173 K. The estimation of Li nuclear-density by maximum entropy-method (MEM) from the neutron diffraction patterns successfully provided the diffusion pathways of Li atom in Li2+2xTiO3+x. The activation energies of the Li diffusion in the pathways were also calculated by nudged elastic band method.

P3-078 **Compatibility of F82H exposed to liquid Pb-Li flow**

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Reduced activation ferritic martensitic steels (RAFMs) are the candidate materials for making blanket module with liquid metal coolant in fusion reactor. Particularly, F82H steel amongst RAFMs has attracted attention because of the excellent properties under neutron irradiation at high temperature below 550°C. However, the compatibility data of F82H in liquid Li-Pb are considerably limited. In the previous test under a practical condition (flow speed: ~0.60m/sec, temp.: ~650°C) with the loop system at Kyoto University, the boundary between F82H and Li-Pb has not shown significant appearance change, while SUS316 has shown the loss of Ni, because chrome oxide scales work as protective layers. The present study investigates the corrosion behavior of F82H in liquid Li-Pb. Using a rotating disk apparatus[1], the effect of flow speed as a function of the radical position on the disk, duration time, and temperature on the corrosion behaviour was evaluated. F82H disks were located at a crucible filled with liquid Li-Pb and rotated at 200rpm (that corresponds to 0.15~0.52 m/sec of apparent flow speed) for a few hundred hour. The high temperature tests at more than 550°C (600°C etc.) were operated for accelerated test. After testing, the cross-section surface of the material was observed by SEM, EDS and EPMA.

As a result, effect of rotating speed for corrosion behaviour of F82H is not significant at 600°C in 100 hour, because high temperature corrosion did not start at this short time. Further results will be reported in the conference.

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P3-079 **Evaluation of multi-layered hardness in ion-irradiated stainless steel by nano-indentation technique**

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Nano-indentation technique has been used to evaluate the irradiation hardening in ion-irradiated materials for simulating the change of mechanical properties under fusion reactor conditions. The available irradiated volume by ion irradiation, however, is small and limited from the specimen surface. Additionally there is a damage gradient in the ion-irradiated layer. These features provide many scientific and technical challenges such as a hardness depth profile, a softer substrate effect and an indentation size effect. To utilize the hardness data of ion-irradiated material obtained by nano-indentation for engineering aspect, a new approach in analysis needs to be developed.

Considering the damage layer to be multi-layers having a different hardness in each layer, a deformation zone by one indent would penetrate several numbers of layers. Here, the hardness, H, by one indent can be described ideally as $H = \Sigma fj Hj$ where fj and Hj are the volume fraction of a deformed zone and a local hardness, respectively. We experimentally evaluated the local hardness by using nano-indentation technique after sectioning the ion-irradiated layer. A modified-316 stainless steel was annealed and then irradiated up to 1 dpa at 250 ^{o}C by 10.5 MeV Fe3+. The damage layer reached to 2.5mm depth from a specimen surface. Sectioning of the damage layer was carried out by using FIB instrument to 0.6, 1.2 and 1.8 mm depth which correspond to the dose of 0.8, 1.4 and 3.4 dpa, respectively. Then nano-indentation test was carried out.

With the assumption of the volume fraction of a deformed zone to be a hemisphere, the local hardness of each layer, H1, H2, H3 and H4 were obtained to be 180, 259, 390 and 408 in kgf/mm2, respectively. This means the local hardness increase with depth, which can be explained by the damage gradient.

P3-080 **Manufacturing and oxidation behaviour of bulk self-passivating tungsten-based alloys**

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Tungsten and tungsten alloys are candidate materials for the first wall (FW) armour of future fusion reactors. However, the use of tungsten implies an important safety concern: in case of a loss-of-coolant accident with air ingress into the reactor vessel, the high temperatures achieved due to the decay heat would lead to fast tungsten oxidation with the release of volatile radioactively activated tungsten oxides. A possible way to prevent tungsten oxidation is the addition of oxide forming alloying elements that form a self-passivating layer at high temperatures in presence of oxygen. In previous works, different ternary tungsten alloys were manufactured via magnetron sputtering (PVD) demonstrating that tungsten thin films of W-Cr-Si and W-Cr-Ti alloys exhibit excellent self-passivating behaviour when exposed to air at temperatures up to 1000ºC. However, thicknesses of several mm are required for the FW armour and thus, the PVD route is not applicable. For this reason, bulk tungsten alloys with tailored composition are being produced by powder metallurgy. In this work, the properties of W-Cr-Si and W-Cr-Ti bulk alloys of different compositions manufactured by mechanical alloying (MA) and hot isostatic pressing (HIP) are shown. After HIP, fully dense samples with extremely fine microstructure are obtained. The oxidation behaviour of bulk alloys was studied by thermogravimetry and in a furnace under synthetic air up to 1000°C. For both thin films and bulk material of the W-Cr-Ti system, a thin Cr2O3 protective layer was found at the outer surface after oxidation at 800 and 1000ºC. The oxidation rate of thin films and bulk materials is compared and discussed. The microstructure after HIP, the existing phases and the resulting self-passivating layers were observed by FEG-SEM, FIB cross sectioning, EDX mapping and X-ray diffraction. Thermal conductivity values and mechanical properties as a function of temperature are presented.

P3-081 **Swelling of SiC materials and its helium effects for expected operating conditions based on some blanket design using SiC materials**

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A silicon carbide fiber-reinforced silicon carbide matrix (SiC/SiC) composite is a promising candidate material for the advanced fusion DEMO blanket because of the excellent themo-mechanical and -chemical properties and irradiation tolerance of SiC itself. Currently, the composites have two options for the DEMO design: materials functionally used such as a flow channel insert in Pb-Li cooling system, and ones used as a part of structural components. Discussing the use of SiC materials for fusion applications, three issues are always coming to arise; 1) to begin with, very few design concept are available for SiC now, 2) environmental conditions for both two purposes are totally different, and 3) more generated transmuted helium is expected and it can affect swelling, which is one of the most important parameters for the designing, compared to other candidate materials. Hence the swelling behavior of SiC with He effects including two target regions needed to be throughoutly and systematically data-based and the soundness of its design should be discussed. Therefore, for the background described above, this paper primarily aims 1) to show examples of Japanese blanket concepts, such as SlimCS and VECTOR, which SiC materials are considered to be used, comparing with other US or EU designs, and 2) to summarize/organize the neutron/ion irradiated swelling data currently obtained for the target temperature ranges of mainly ~700°C for functional use and ~1000°C for structural, with a strong emphasis on the effects of helium on swelling of SiC.

P3-082 **IFMIF-Test Facilities: Functional Analysis and Improvement of Hot Cells**

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

Alongside, also the arrangement of the cells to provide fast and high efficient work on the modules is described in this paper.

P3-084 **MYRRHA a Flexible And Fast Spectrum Irradiation Facility for Fusion Reactor Materials Testing**

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Since 1998, the Belgium Nuclear Research Centre is developing a multipurpose irradiation facility in order to support research programs on fission and fusion reactor structural materials and nuclear fuel development. MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) is a flexible experimental accelerator driven system (ADS) with a thermal power of 100MWth, able to work in both subcritical and critical mode. The Belgian federal government has approved early 2010 the funding of this international project, which from 2023 onwards, will contribute to the development of innovative solutions in the field of nuclear technologies. One of the objectives of this new research reactor is the ability to test and validate materials for GEN IV systems and fusion reactors. Therefore the MYRRHA reactor is conceived with a fast core, cooled by lead-bismuth, and with a 600MeV proton beam, generating extra neutrons in the core by spallation. MYRRHA can load 7 experiments, on 37 positions. The performances at the central positions are high enough to generate the dpa/y rate for fusion experiments. Although the He/dpa ratio is significantly higher than in thermal spectrum MTR's, it does not reach the He-dpa levels for the first wall. Such positions inside of the core region are however interesting to simulate the conditions at and behind the breeding zone. Specifically in sub-critical mode (with proton beam), the central peak flux is boosted to values which can almost simulate the conditions at the first wall of the fusion reactor, in terms of dpa/y and He/dpa. Temperature controlled experiments can be loaded below the beam window, to allow testing under representative conditions.

The paper will give the forecasted neutronic and protonic irradiation conditions, the calculated dpa and He production rates as well as a first conceptual design of the specific irradiation facility for fusion material at high He/dpa ratio. The part of MYRRHA which support fusion is called MYRRHA – IMIFF (Innovative Material Irradiation Facility for Fusion). Its capabilities will be shown in the paper.

P3-085 **The development at a pilot plant scale and characterization of a reduced activation ferritic-martensitic steel for fusion applications, Asturfer®**

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The optimization and validation of materials to be used in the ITER TBM modules is a main interest in fusion power development. Reduced activation ferritic-martensitic (RAFM) steels, in particular the Eurofer grade, are considered promising candidates as structural material for the EU reference concepts. Eurofer is a 9CrWVTa RAFM steel, which exhibits a stable tempered martensite microstructure that is intended to allow operation up to 550 ºC.

This paper shows the work carried out in the frame of the design, development, and metallurgical process optimization for the development of a new experimental RAFM steel at a pilot plant scale with a chemical composition and mechanical properties that fulfill the Eurofer specifications. The experimental RAFM steel grade was obtained in a high Vacuum Induction Melting furnace in order to assure a good control of the chemical composition and to avoid possible impurities. The ingot, with a thickness of 100 mm, was hot rolled to a final thickness of 14 mm and finally heat treated. The whole manufacture process performed is showed in detail as well as the evaluation of the main microstructural features and mechanical properties: tensile strength up to 550 ºC and impact toughness. The new steel meets the requirements concerning chemical composition, microstructure and mechanical properties. In comparison with Eurofer, it must be highlighted that, despite its lower USE, the DBTT of Asturfer® is lower than that of Eurofer, which means a longer in-service life.

This work has been supported by the Spanish Ministry of Science and Innovation under the CONSOLIDER TECNO-FUS INGENIO 2010

P3-086 **Diffusion bonding for 9Cr-ODS and JLF-1 Reduced Activation Ferritic/Martensitic Steels**

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Bonding technique for oxide-dispersion-strengthened (ODS) steels is essential for its application to the blanket system with reduced activation ferritic/martensitic (RAFM) steels. The present study fabricated diffusion bonding joints of the candidate 9Cr-ODS steel and JLF-1 RAFM steel by hot isostatic pressing (HIP) process. The bonding strength was evaluated to obtain the optimum bonding condition. Materials used were 9Cr-ODS steel with a chemical composition of Fe- 9.08Cr- 0.14C- 1.97W- 0.23Ti- 0.29Y- 0.16O- 0.013N and JLF-1 JOYO heat with Fe- 9.00Cr- 0.090C- 1.98W- 0.20V- 0.083Ta- 0.015N. The specimen size of 9Cr-ODS disk was 5 mm in thickness and 24 mm in diameter, while JLF-1 block was 20 mm in thickness and 24 mm in diameter. The ODS disk was sandwiched with 2 JLF-1 blocks and canned into a soft steel capsule by electron beam welding. The capsule was HIPed at 1273 for 3 hr under 191 MPa. Since the HIPing temperature was higher than the tempering temperature for 9Cr-ODS, 1073 K, and for JLF-1, 1053 K, isochronal annealing was examined to recover their hardness to the level before the bonding. Impact specimens were machined from the HIPed and annealed joint. V-notch was placed on the boundary between 9Cr-ODS and JLF-1 to evaluate the bonding strength. Impact properties of the joints and optimum annealing condition after the bonding will be discussed.

P3-087 **Optical absorption defects created in SiO2 by O, Si and He ion irradiation**

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Fused silica is a key element for optical components in magnetic and inertial fusion devices, and these components will be within a radiation field. Optical and electrical properties of silica are strongly influenced by defects, which are introduced during the manufacturing process or produced by energetic photons (UV light, X-ray or γ ray) and/or particles (ions, electrons or neutrons). To emulate the neutron irradiation damage, it is known that ion implantation is widely used as a relatively low-cost and rapid means, without activation, of introducing radiation damage in materials, and at the same time, studying the damage irradiation by alpha particles coming from plasma. In this work high purity fused silica with different OH content: KU1 (high OH) and KS-4V (low OH), highly radiation resistant and considered as candidate materials in ITER, were irradiated with silicon, oxygen and helium ions at several fluences (from 5e12 to 1.6e15 ions/cm2). A commercial silica (Infrasil 301) with higher metallic impurity content was also used for comparison. The energy of the ions was: 24.37 MeV Si+4 ions, 13.5 MeV O+4 ions and 2.455 MeV He+ ions. Depth profiles of ions energy loss were simulated using SRIM code. The irradiations produce mainly damage by electronic excitation and the estimated depth was ~ 9 µm for the three ions. After ion implantation the absorption spectra was measured in a wide wavelength range, from vacuum ultraviolet (VUV) to infrared (IR) and the behaviour of defects (as E´, ODC, NBOHC, POR) created by the irradiation with ions at different doses has been analysed.

P3-088 **CFD analysis on the effect of the flow straightener of IFMIF/EVEDA Lithium Test Loop**

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To strive for the realization of the International Fusion Materials Irradiation Facility (IFMIF), the Engineering Validation and Engineering Design Activities (IFMIF/EVEDA) project is under progress. One of the key validation facilities is the EVEDA lithium-testing loop, which has been constructed in the Oarai Research and Development Center, JAEA, in Nov. 2010 and is under operation. The lithium velocity in the 6 inch piping upstream of the target assembly is up to around 2 m/s. The various bends in the piping are expected to create an inhomogeneous flow velocity distribution, which would affect the free surface flow in the target assembly. A flow straightener, consisting of a perforated brick and three perforated plates, is therefore installed above the reducer nozzle. In order to evaluate the influence the non-uniform flow velocity distribution to the target flow and the efficiency of the flow straightener, Computational Fluid Dynamics (CFD) analysis has been carried out by ANSYS-FLUENT from the piping to the nozzle entrance. The CFD calculations show that the flow velocity variation in the cross section downstream of the 3rd perforated plate perpendicular is rectified to about 3.6% of mean velocity compared to 22.7% at the inlet of the flow straightener, in case the IFMIF operation flow velocity condition of 15 m/s in nozzle. Additionally, it turned out that the turbulence intensity is decreased to about 4 to 5 % from 19.5% by the flow straightener.

P3-089 **Joining Techniques for Reduced Activation 12Cr Steel for Laser Inertial Fusion Energy**

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A reduced activation version of the ferritic martensitic steel HT9 is being developed as a potential chamber wall material for Laser Inertial Fusion Energy (LIFE). To join these candidate material components, a variety of conventional and solid-state welding processes are being considered. One promising technique, hot isostatic pressing, has not yet been attempted with this material. Therefore, diffusion bonds were developed in the reduced activation HT9 steel by joining samples at temperatures between 950 C and 1150 C while under a pressure of 103 MPa for two hours. Specimens extracted from these cans were then characterized to determine appropriate bonding conditions and resulting strength.

Standard welding processes such as electron beam, tungsten inert gas, and laser welding were also performed to join the steel. After welding, the specimens were normalized between 750 C and 1050 C to determine appropriate post-weld processing. To isolate the ideal normalization temperatures, we analyzed the welds using nano-indentation, electron backscatter diffraction, and electron microscopy. The changes in microstructure of the fusion and heat-affected zones after normalization demonstrated the necessity for post-weld heat treatment. Finally, the mechanical quality of the different weld processes was determined by performing tensile and bending strength of the normalized and as-welded conditions.

Findings from this work will guide weld process selection for fabrication design. However, further characterization will be necessary to qualify these weld processes for use, particularly if the welds will be located in the high-fluence regions of the fusion system.

P3-090 **Assessment of the beam-target interaction of IFMIF: a state of the art**

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One of the key issues of the IFMIF/EVEDA project is to demonstrate a stable free surface flow of the Li jet in the concave flow channel of the target assembly. This validation is done experimentally in the EVEDA Li test loop built in the JAEA Oarai lab and is accompanied by CFD calculations of Japanese and European partners. The waviness of the jet must stay within rather narrow limits to protect the backwall from beam impingement effects and to maintain stable irradiation conditions in the test modules. The validation, however, is purely hydraulic under iso-thermal conditions. In IFMIF, interaction of the high power deuteron beam with the Li target needs to be taken into account. Thermal and momentum transfer of the beam may destabilize the flow structure, cause shock waves and increased evaporation or aerosol formation. Different aspects of beam interaction have been analyzed in the past, but a comprehensive assessment is still lacking. The contribution will provide an overview of the IFMIF related beam-free Li surface interaction studies and re-assess the effects for two conditions: pulsed beam operation during start-up of the accelerator and high-speed flow of the Li across the beam footprint with a flat-top profile.

P3-091 **Microstructure and deuterium permeation of alumina coatings on CLF-1 via MOCVD**

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Tritium permeation through structural materials in the first wall and blanket is a key issue in the development of future fusion reactors. Alumina is a promising tritium permission barrier material to reduce the tritium permeation rate by several orders of magnitude. In this work, alumina coatings were deposited by MOCVDon CLF-1,which isa newly developed Chinese RAFM steel. The coating was characterized by X-ray diffraction (XRD), scanning electron microscope (SEM) and X-ray photoelectron spectroscopy(XPS). It was found that the alumina coating was amorphous. The alumina coating had a dense and uniform microstructure. The hydrogen permeation inhibition performance of coatings was investigated by deuterium permeation experiment at 823-973K.The alumina coatings showed effective hydrogen permeation suppression performance on CLF-1.

Topic F Neutronics

P3-092 **Investigation of the requirement for an energy storage system for DEMO and FPP**

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The recent consideration of the development towards a pulsed DEMO device brought up the question of energy storage requirement for both DEMO and a future Fusion Power Plant. Therefore an investigation was initiated in this subject in the frame of EFDA to establish contacts with grid operators around Europe; to collect and compare the recent and planed future requirements of the grids in terms of limitation on maximum rate of change of active power for start-up and ramp-down phases of the fusion device; to define requirements for an energy storage system for a pulsed DEMO and FPP; and finally to explore and assess potential energy storage technologies. The present paper intends to introduce the achieved results, in particular showing how the identified requirements influence the energy storage needs of a fusion power plant and how this differs between countries. It will also describe on what basis the three most promising technologies (pumped thermal electricity; cryogenic liquid; molten salt) are selected from a range and the results of their further detailed analysis.

P3-093 **Nuclear Analysis of the IFMIF European Lithium Target Assembly System**

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In the framework of the current Engineering Validation and Engineering Design Activities (EVEDA) phase of the International Fusion Materials Irradiation Facility (IFMIF) project, ENEA is in charge of the design of the European version of the target assembly system which employs a removable bayonet backplate concept. The latter, alternative to the Japanese concept based on an integrated solution, allows for the possibility of periodically replacing only the most irradiated and thus critical component (i.e., the backplate) while continuing to operate the rest of the target for a longer period. In the latest years, significant progresses have been made in the development of the target design and its status is now in a well advanced stage. With the objective of assessing the nuclear behaviour of the system and supplying the necessary input data to the thermomechanical and thermohydraulic calculations, three different types of analysis have been performed: neutron-gamma transport calculations, neutron activation calculations and power deposition calculations in the lithium jet. Neutron-gamma transport calculations have been carried out using the MCNP5 1.6 code integrated with the McDeLicious-11 neutron source provided by KIT. A strong effort was made to suitably rebuild the 3D CAD model of the target system in a such a way it could be properly imported in MCNP through the McCad/MCAM interface software. Neutron activation calculations have been performed by means of the EASY-2010 activation code package in order to provide radioactive inventories for decay heat evaluation and safety analyses. The power deposition profiles in the lithium have been calculated via MCUNED code considering the real 2D spatial variation of the beam intensity in the footprint area.

This paper presents the main results obtained from the above analyses including, among others, neutron and gamma fluxes, heat deposition, radiation damage and gas production in the whole target assembly.

P3-094 **Benchmarking of MCNPX Results with Measured Tritium Production Rate and Neutron Flux at the Mock-Up of EU TBM (HCPB concept)**

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Europe is currently developing two reference breeder blankets concepts for DEMO reactor specifications that will be tested in ITER under the form of Helium-Cooled-Lithium-Lead (HCLL) and Helium-Cooled-Pebble-Bed (HCPB).
In order to reassesses the available results a framework contract agreement between F4E and IDOM (Spain)has been signed. Shielding Engineering and Analysis (SEA-Spain) and Universidad Nacional de Educacion a Distancia (UNED-Spain) participate as sub-contractors of IDOM. In this study, a qualification of MCNPX code and nuclear data libraries are benchmarked with measured tritium production rate and neutron flux at the mock-up of the EU TBM, HCPB concept. The irradiation and measurements were performed in the frame of European Fusion Technology Program by ENEA (Italy), TUD (Germany) and JAERI (Japan).
The mock-up consisted of three metallic beryllium blocks and two double cassettes of Li2CO3 ceramic breeder material. The mock-up is provided with four penetrations at 4.2, 10.5, 16.8 and 23.1cm from the mock-up front surface. These penetrations were used to locate detectors for measuring the tritium production in the ceramic breeder and the neutron flux in beryllium material. Stacks of 12 Li2CO3 capsules corresponding to double cassettes of ceramic breeder with four depths were used during the irradiation for tritium production rate measurements. The neutron flux during the irradiation was measured at different depths in the central beryllium block by using the activation of Nb93(n,2n), Al27(n,a) and Ni58(n,p).

The calculations are performed with MCNPX code using a three dimensional modelling of TBM mock-up including Li2CO3 cassettes and four penetration channels. The detection capsules for tritium production rate and the activation foils for neutron flux calculations are modelled with full measurement detail. The D-T 14MeV neutron spectrum and corresponding angular distriburion is used.

P3-095 **Start-Up and Shutdown Thermomechanical Transient Analyses of the IFMIF European Lithium Target System**

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IFMIF (International Fusion Materials Irradiation Facility) is a high-flux neutron irradiation source which is currently being designed with the aim to provide the fusion community with a machine for testing candidate materials to be used in future fusion power reactors. In the framework of the current IFMIF Engineering Validation and Engineering Design Activities (IFMIF/EVEDA) phase, ENEA is responsible for the design of the European concept of the IFMIF lithium target system which foresees the possibility to periodically replace only the most irradiated and thus critical component (i.e., the backplate) while continuing to operate the rest of the target for a longer period (bayonet backplate concept). In the latest years, significant progresses have been made in the development of the IFMIF target design and its status is now in a well advanced stage. With the objective of evaluating the performances of the system in terms of temperature, stress and displacement fields evolution during start-up and shutdown phases, an uncoupled thermomechanical transient analysis has been performed in close collaboration with the University of Palermo by means of a qualified finite element thermomechanical code. The calculations employed a realistic 3D time-dependent FEM model which takes into account all the mechanical and thermal loads including the nuclear heating due to neutron and prompt gamma fields during start-up and decay power of activated products during shutdown. The nuclear data have been calculated by ENEA as part of an extensive neutronic analysis carried out through the MCNP transport code and the EASY-2010 activation code package and then passed as input to the thermomechanical FEM model. In this paper, the results of the above thermomechanical transient analyses carried out under nominal conditions are reported, highlighting the relevant indications obtained with respect to the fulfillment of the design requirements and possible hints for improving the system design.

P3-097 **Challenges and Progress in the Design of ITER Tokamak Cooling Water System Inside Cryostat**

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The ITER Tokamak Cooling Water System (TCWS) has the main function to provide the cooling of the Plasma Facing Components (PFCs) and the Vacuum Vessel (VV) and to transfer their heat loads to the atmosphere via Component Cooling Water and Heat Rejection Systems. The TCWS is designed to remove the total peak heat load of about 1 GW and is divided in three Primary Heat Transfer Systems (PHTSs), two Chemical and Volume Control Systems (CVCSs), a Draining and Refilling System (DRS) and a Drying System (DYS). The PHTSs are three separated circuits with independent thermal hydraulic functions for cooling of the Vacuum Vessel (VV PHTS), the Blanket Divertor, the in-Vessel coils and the Equatorial Ports Clients (IBED PHTS), and the Neutral Beam Injectors (NBI PHTS). The water chemistry parameters of these three PHTSs are precisely maintained at three different levels based on the requirements of the system. The TCWS has the important safety role of providing primary confinement of radioactivity. Radioactive materials in the cooling water are Activated Corrosion Products (ACP) and Tritium, which permeates the water through the Plasma Facing Components (PFC)s. The VV PHTS removes the decay heat from the VV and PFCs even when the other PHTSs are not available during the off-normal accidental events like the Loss of Cooling Accident (LOCA) and the Loss of Site Power (LOSP). The TCWS design is now in its final stage, having completed the definition of physical and functional interfaces with the system clients,

the selection of the piping layout and its arrangement and routing inside the secondary confinement, the definition of the pipe supports and the penetrations inside the Tokamak building. ITER-IO has directly performed a specific design of the TCWS piping routing and penetration from the Bioshield inside the Cryostat. This design has specifically addressed all the safety requirements for protecting the pipes with barriers against leaks of radioactive water. The piping stess analyses have been performed according to the ASME B31.3 for the forseen load combinations that consider thermal, seismic, magnetic and other typical dynamic events dictated by the Plasma fast transients and displacements inside the VV. The paper describes the main challenges and the specific design progress carried-out for the TCWS piping layout and stress analysis inside the Cryostat.

P3-098 **Detailed 3-D Nuclear Analysis of ITER Blanket Modules for Final Design Review**

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In ITER, the blanket module (BM) main components are the First Wall (FW) panel and Shield Block (SB). The FW panel consists of Be armor, Cu heat sink, and steel structure with embedded water coolant. The SB consists of steel structure with embedded water coolant. The design process for the BM includes assessment of thermal stress, detailed computational fluid dynamics (CFD), and electromagnetic (EM) analyses. Re-welding is required at several locations in the BM and the VV behind it and this requires accurate determination of helium production in the structural material.

Therefore, detailed mapping of nuclear heating, radiation damage, and helium production is an essential input to the design process. Additionally, several key ITER components are located adjacent to, or within the BMs such as the ELM and VS coils, and the NBI ports. These components also require nuclear analysis for their design process. Because the BMs, ELM coils, and NBI ports are geometrically complex, a CAD based approach to neutronics is ideal for generating detailed nuclear radiation parameters. In this work we will use the CAD based DAG-MCNP5 transport code to analyze detailed models inserted into a 40 degree partially homogenized ITER model. The models used will be the most up to date models available at the time of the blanket final design review (April 2012). These models will incorporate the changes to the ITER design since the time of the blanket preliminary design review. These design changes include such items as thickening of the inboard BMs, changes to the poloidal manifolds, and changes to the In-Vessel coils and feeders.

Detailed maps of nuclear heating, radiation damage in terms of atomic displacement (DPA), and helium production will be presented. Comparisons to previous CAD based analysis will also be presented. These results are actively being used to guide the BM design process.

P3-100 **Analysis of radiation environment at divertor in helical DEMO reactor FFHR-d1**

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In helical reactors, magnetic field lines generated by helical coils can be extracted from core plasma to back side of radiation shield without crossing first walls of breeding blanket. This indicates that divertor components could be placed behind radiation shield to suppress radiation damages and radioactivity.

Since all breeding blanket, radiation shield and divertor have helical structures along helical coils, neutron streaming through openings between the components must be evaluated accurately to analyze radiation environment around divertor components placed behind radiation shield. Impact of divertor port openings on radiation shielding for helical and poloidal coils is also significantly important issue to be evaluated accurately. The present study has been conducted by improving a 3-D non-axisymmetric calculation model of FFHR-d1 for the Monte Carlo transport code MCNP5.

Calculated distribution of fast neutron flux of >0.1 MeV in FFHR-d1 shows that the magnitude of the flux behind radiation shield is ~4 x 10^12 n/cm2/s for the averaged neutron wall loading of 1.5 MW/m2. The value is lower by more than one order compared with ~1-4 x 10^14 n/cm2/s at first walls facing to core plasma. In the FFHR-d1 design, a large space is provided around divertor components placed behind radiation shield layer. The magnitude of fast neutron flux could be suppressed to half by installing additional shield for divertor.

These results indicate that damages on copper can be <~1 dpa after operation of several years under the moderate neutron wall loading of FFHR-d1 and the above efforts in positioning and radiation shielding. Since degradation in the elongation property is suppressed, copper materials could be adopted for cooling channels of the divertor. Detailed analyses of irradiation damages on copper materials and impact on radioactivation are being performed at present.

P3- 101 Novel Hybrid Monte Carlo/Deterministic Technique for Shutdown Dose Rate Analysis

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Accurate predictions of dose rates from activated structural materials during shutdown — referred to as shutdown dose rate (SDDR) — are necessary to support operation, maintenance, and waste disposal planning for fusion reactors. The rigorous 2-step (R2S) method consists of Monte Carlo neutron and photon transport calculations coupled to a comprehensive activation step with a dedicated inventory code and library. Accurate full-scale Monte Carlo calculations of the R2S method are impractical for fusion reactors because they require calculating space- and energy-dependent neutron fluxes everywhere inside the reactor. The use of global Monte Carlo variance reduction techniques was suggested for accelerating the neutron transport calculation of the R2S method. The prohibitive computational costs of these approaches, which increase with the overall problem size and amount of shielding materials, inhibit their use in the accurate full-scale neutronics analyses of fusion reactors. This paper describes a novel hybrid Monte Carlo/deterministic technique that uses the Consistent Adjoint Driven Importance Sampling (CADIS) methodology but focuses on multi-step shielding calculations such as the R2S calculations of SDDR. This technique, referred to as Multi-Step CADIS (MS-CADIS), speeds up the Monte Carlo neutron calculation of the R2S method using an importance function that represents the importance of the neutrons to the final SDDR. Using a simplified example, preliminarily results showed that the use of MS CADIS enhanced the efficiency of the neutron Monte Carlo simulation of an SDDR calculation by a factor of greater than 550 compared to standard global variance reduction techniques, and that the increase over analog Monte Carlo is higher than 10,000. Application of this hybrid technique to realistic fusion problems and development of new methods for propagating the uncertainties from the photon source to the final SDDR are currently underway.

P3-102 **Primary Design of EAST RMP Coils**

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During the seventh plasma discharging experiment of EAST in 2012, the stationary and repeatable H-mode was achieved. Aiming to improve the plasma control, the EAST resonant magnetic perturbation (RMP) coils are under design, which integrate the functions of error field correction, edge localized mode and resistive wall mode.

This paper will introduce the primary design of EAST RMP coil. The RMP coils consist of 8 upper saddle coils and 8 lower saddle coils. The upper and lower coils are designed to allow operation with AC currents up to 12kAturn, f = 50Hz for ELM control, and current 1kAturn, f = 1 kHz for RWM feedback stabilization. The coils are designed as four turns with 3kA current. Considering the EAST VV would baking to 350 degree, the RMP coils use ITER-type conductor, which consist of copper conductor with active cooling water 3mm MgO insulation layer and 2mm SS316L Jacket. The total dimension of conductor is 28mm, and the cooling channel is 10mm with 4m/s water in it. The relevant Electromagnetic and structural analysis has been done. Keywords: EAST tokamak, RMP coils, design

P3-103 **Nuclear Analysis of the Diagnostics Equatorial Port Plug #3 In ITER with Attila Code and Impact on Interspace Dose Rates**

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The equatorial port #3 (EPP3) in ITER will house the Motional Stark Effect, MSE, and the Charge Exchange Recombination Spectroscopy, CXRS. The presence of these diagnostics leads to inter mangling labyrinths through which neutrons and gamma rays finds their ways to the closure back plate and hence raise the decay dose rate inside the ports interspace areas where maintenance is performed. The design of the plugs calls for maintaining a dose level in these areas not to exceed 100 mSv/hr 106 sec (~11 days) after shutdown.

In the present study the interspace dose rate assessment follows this approach: (1) estimate the interspace dose rate in a baseline EPP3 port plug prior to installing any diagnostics. In this regard, the EPP3 is the only port considered in the calculation model used with no account to any dose rate contribution from the activation of the upper and lower port plugs. In the methodology adopted, about half of the 100 Sv/hr dose rate limit is from neutron not associated with the diagnostics or the vertical shield modules, (2) estimate the increase in the interspace dose rate upon installing the diagnostics. Each of the three vertical shield modules in the EPP are allocated a dose rate budget of 17 Sv/hr above the baseline, and (3) introduce some design changes if the dose rate above the baseline exceeds the allowed budget. The EPP3 designs studied, are: (1) 2-view CXRS and MSE installed with no back shield, (2) same but with back shield, and (3) 3-view CXRS and MSE with back shield. The later design resulted in the least average dose rate 1-m away from the closure back plate. Nevertheless, the resultant dose rate is an order of magnitude higher than the allowed budget of ~34 Sv/hr (two diagnostics). This is mainly due to the relatively large cross section area of the Z-shaped dogleg labyrinth used in the design. The results of a parametric analysis for the neutron flux dependence at the back plate on the dimensions/location of Z-shaped dogleg, will be discussed in the paper. The optimized parameters were used as guidance to minimize the interspace dose rates. All the transport and dose calculations were performed with the attila 3-D discrete ordinates code along with the FENDL-2.1 data base.

P3-104 **Design and Configuration Management Platform for Fusion Components**

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Fusion components development, as actively cooled plasma facing components (PFC), required the use of many different design and analysis tools from research and development to operation phase. These components, often developed within the frame of collaborations involving labs and industries, could be designed through different native Computer-Aided-Design formats. Moreover, during the development phase, many iterations are necessary between analysis and design using specific 3D models depending on the kind of analysis (mechanical, optical, photonic, physics, neutronic…). To enable an efficient development process, a design and configuration management platform could be used to manage the data workflow, the reference design (also called configuration) as well as its interfaces with the relevant components. All the different 3D representations for design or analysis as well as the specification or interface documents must be linked to a reference component and follow a lifecycle workflow. These data must be centralized and reachable by all the different codes involved in the development process in order to guarantee the analysis results consistency. This paper presents the technical solutions that are used and under development for managing the lifecycle design of the WEST project components. It describes the approach to implement a data and configuration management platform and focuses on the way to manage in the same digital mock-up all the 3D representations dedicated to design, simulation or analysis. The paper explains the interfaces that are implemented or in construction to connect together the different tools like Documents Management System, CAD modeller, multi-physics or optical analysis codes around the data management backbone. Tools which are already in service have led to a reduction by a factor 6 of the time for 3D CAD data simplification used as input for 3D predictive photonic modelling. Finally the paper presents the methodologies and workflows used on the WEST project to optimize the lifecycle design and finally contribute to reach the project targets in terms of performance, cost and schedule.

Topic G Safety Issues

P3-105 **IFMIF Accelerator Facility Safety Analysis Based on FMECA Methodology**

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Present paper shows IFMIF Accelerator Facility safety analysis performed, based on Failure Mode, Effects and Criticality Analysis (FMECA) tool, and in the frame IFMIF–Tractebel contract. FMECA is performed as iterative work between engineering design and safety requirements Taking into account that main safety principle assumed in IFMIF is defense in depth, FMECA analysis provide qualitative and quantitative results on passive engineering control (confinement barriers) to reduce the risk towards minor risk and to figure up remaining potential open questions at intermediate design step.

FMECA Methodology follows referential safety documents produced by IFMIF safety working team, i.e. “hazards evaluation guideline” and “safety specifications document”. It is based on 4x5 risk matrix, ranking frequency in 5 rating scale from extremely unlikely to normal operation and Severity as negligible, marginal, critical or catastrophic. Target considered, as the most restrictive at plant level, in safety analysis was the worker.

Initially five configuration states were defined: 1. Irradiation (normal operation) where two options were studied: (a) two beams in operation and (b) one beam in operation and one beam under maintenance, 2. Maintenance – start-up, 3. Maintenance-shutdown, 4. Maintenance, 5. Decommissioning. Analysis effort is focused on Irradiation configuration - including the two normal operation modes (a) and (b)- and the particularities in case of two beams are in maintenance (start-up and shout down).

Current results of FMECA analysis are discussed in present paper showing that in normal operation modes, the technological barriers and MPS (Machine Protection System) are sufficient to reach minor risk to every AF components. Additionally maintenance modes are shown to be the most critical configuration state for AF. Nevertheless, combined failures were not studied.

Finally, from FMECA analysis, the accelerator facility safety requirements were refined and internal Potential Initiating Events (PIEs) evaluated. Resulting PIEs are discussed at present developing stage. Authors acknowledge Spanish Ministry of Economy and Competitiveness by financing “PROYECTO IFMIF-EVEDA (II), Programa Nacional de Internacionalización de la I+D, REFERENCE: AIC-A-2011-0654”

P3-106 **Free License codes to simulate the diffusion of contaminants in case of radiological release**

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The radiological risk is connected to a wide range of activities, beginning with the medical and military ones and including those connected to the industrial and research activities such as nuclear fusion. A valid tool to predict the consequences of the accidents and reduce the risk is represented by computing systems that allow modeling the evolution of a possible release of radioactive materials over time and space. In addition to proprietary codes there are free license codes, like Hot-Spot, that allow providing a set of tools to simulate diffusion in case of accidents involving radioactive materials and analyze the safety and security of the facilities in which the radioactive material is manipulated. In this paper, the authors simulate, by means of the Hot-Spot code, an accident of a plant for reprocessing radioactive fuel and compare the numerical data with experimental ones measured in-situ and published by the IAEA in the report "The radiological accident in the reprocessing plant at Tomsk". The code, validated with data measured in situ, has been used to simulate a diffusion of radiological contaminants in a nuclear fusion experiment and the results are presented. The aim of this work is to demonstrate the capability of free license codes to model the radiological diffusion in case of accidents in order to guarantee the safety of people and operators and the security of the plants. Both are critical issues for the construction of nuclear fusion experiments like ITER.

P3-107 **Dust tracking techniques applied at STARDUST facility: first results**

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An important issue related to future nuclear fusion reactors fueled with deuterium and tritium is the creation of large amounts of dust due to several mechanisms (Disruptions, ELMs, VDEs) responsible for its generation. The dust size expected in the nuclear fusion experiments (such as ITER) is of the order of microns (between 1 and 100 μm). Almost the total amount of this dust remains in the Vacuum Vessel (VV). This radiological dust can re-suspend in case of LOVA (Loss of Vacuum Accident) and these phenomena can cause explosions and serious damages to the health of the operators and to the safety of the device. The authors have developed a facility, STARDUST, in order to reproduce the thermo-fluidodynamic conditions comparable to those expected inside the VV of the next generation of experiments such as ITER in case of LOVA. The dust used inside the STARDUST facility presents particle sizes and physical characteristics comparable with those that created inside the VV of nuclear fusion experiments. In this facility an experimental campaign has been conducted with the purpose of tracking the dust re-suspended at low pressurization rates (comparable to those expected in case of LOVA in ITER and suggested by the General Safety and Security Report ITER-GSSR) using a fast camera with a frame rate from 1000 to 10000 images per second. The velocity fields of the mobilized dust derived from the imaging of a two-dimensional slice of the flow illuminated by a laser beam. The aim of this work is to demonstrate the possibility of in dust tracking by means of image processing with the objective of determining the velocity field values of dust re-suspended during a LOVA.

Keywords: *Nuclear fusion plants, Security, Safety, Re-suspension, Dust Tracking, Radioactive, Computer Vision, Particle Image Velocimetry (PIV)*

P3-**108 Neutronics analysis and dose reduction for the ITER Neutral Beam Cell**

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In ITER the three heating neutral beam ducts are large penetrations through the bioshield which leads to significant levels of neutron streaming and hence activation in the ITER Neutral Beam (NB) Cell. Unlike many of the penetrations through the bioshield it is not possible to reduce the size or introduce shielding as this would interfere with the operation of the beams.

Extensive neutronics analysis simulations using MCNP5 and FISPACT have been performed to calculate both the on load and shutdown dose rate in and around the NB Cell. A large MCNP model was developed coupling the already existing 80 degree sector model of ITER with updated detailed models of the neutral beam injectors, remote handling equipment and other bulky items present in the NB cell and in the L3 area where the transmission lines pass through. The simulated neutron spectrum results are coupled to FISPACT using the MCR2S software to produce a shutdown photon source using the rigorous two step method. This shutdown photon source is then used to calculate the shutdown photon dose rate at a given time after shutdown.

A series of simulations were undertaken focusing on a single area such as a potential lead shield wall or around the penetration between the NB cell and the L3 area. In each area improvements to the design were suggested in order to reduce the dose rate. The overall effect has been to reduce the dose rate in key locations for during plasma operations and during shutdown and hence help to demonstrate that the design is in line with the ALARA principle.

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P3-109 **Numerical Study of Air Jet Flow Field During a Loss of Vacuum Accident inside STARDUST facility**

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Air leakage into tokamaks vacuum vessel during plasma burning or maintenance operations may lead to their pressurization with relevant consequences both on the operations and workers safety. A preliminary analysis of the physical phenomena involved in such accidents is necessary in order to study the thermal-fluid dynamics after air ingress into the vacuum vessel. The numerical simulation of Loss Of Vacuum Accident (LOVA) scenarios is a challenging task for today numerical methods and models because it involves large volumes, air flows ranging from highly supersonic to nearly incompressible and contemporary heat transfer. Accuracy of the numerical results is also required in order to provide a sufficient margin in the design of the safety systems. In this contribution, the authors present and discuss the results of numerical simulations of air jet flow field during a LOVA with particular attention to the comparison with the experimental data and differences arising from the use of different types of grid resolution and turbulence models (Zero-Equation, k-ω and SST).

P3-110 **Study of safety features and accident scenarios in a fusion DEMO reactor**

Nakamura, Makoto[1]; Tobita, Kenji[1]; Gulden, Werner[2]; Sakamoto, Yoshiteru[1]; Someya, Youji[1]; Tanigawa, Hisashi[1]; Araki, Takao[3]; Matsumiya, Hisato[3]; Ishii, Kyoko[3]; Utoh, Hiroyasu[1]; Takase, Haruhiko[4]; Hayashi, Takumi[1]; Sato, Akira[1]; Yonomoto, Taisuke[1]; Federici, Gianfranco[2]; Okano, Kunihiko[4]

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It is generally important in ensuring safety of a nuclear machine to demonstrate that (1) nuclear reactions can be terminated, (2) residual heat can be removed and (3) radioactive material can be confined. While any fusion reactors can inherently ensure the former one because “runaway” nuclear reactions are not possible, whether the latter two are demonstrated depends on a reactor design, i.e. the latter two are not always demonstrated in a fusion DEMO reactor. Purposes of this study are to identify safety features in a DEMO reactor and accident scenarios that potentially jeopardize its safety. For these purposes we compare source terms and energies that can mobilize radioactive materials in DEMO with those in ITER and other power plant conceptual designs. We also discuss DEMO design principles to reduce such source terms and energies. Effects of active and passive prevention and mitigation systems in an accident condition are studied by using a thermohydraulics computer code. In order to assess the level of inherent safety features of DEMO, we particularly analyze “bounding scenarios”, i.e. “far” beyond design basis accidents that have an extremely small likelihood of occurrence but potentially expose the large amount of the energies. A specific feature of our study is that in order to enhance inherent, passive safety of the DEMO reactor, countermeasure systems are newly proposed for preventing or mitigating even the bounding accidents. We will study several prevention/mitigation systems, e.g. pressure suppression systems of a blanket module, vacuum vessel and/or gallery, and cooling systems using natural circulations, and etc.

P3-111 **Waste management scenario in the hot cell and waste storage for DEMO**

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Previously, waste management of strategy of fusion reactors has been considered for the after the time of decommissioning. However, radioactive waste is also generated in the periodic replacement of in-vessel component such as blanket modules and divertor. Rather, the management strategy of radioactive waste generated in the periodic replacement may be much more important in the point of view of fusion reactor design, because it has a large impact on the design of the hot cell and waste storage. In the replacement period of a fusion reactor, blanket and divertor modules should be removed from the reactor as an assembly for plant availability. In the hot cell, the modules will be removed from the backplate of the assembly. Since the backplate made of RAFM can be reused, the decay heat must be removed using active cooling to keep the temperature below 550 ^{O}C for structural strength of RAFM.. At the same time, the active cooling must not cause a contamination of the hot cell environment due to blowoff of tritium and tungsten dust. The cooling scenario is one of key points in the waste management. The other point is recycling scenario for rare or useful metal like beryllium and lithium used as multipliers and breeder in the blanket. After the decontamination of the blanket has been completed, Li2TiO3 and Be12Ti pebbles need to be collected for recycling after a certain period of cooling of radioactivity. Since the structural material (RAFM) of the blanket and divertor is not reused nor recycled because of high contact dose rate, the RAFM should be kept in the interim storage until the time of disposal. The problem is the space for storage. Breaking up the RAFM into small pieces reduces the volume of the waste, contributing to a reduction of the storage space.

P3-112 **Radioprotection design of the water cooling system of the LIPAc beam dump**

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LIPAc is a prototype accelerator facility for IFMIF. LIPAc deuteron beam is stopped in a beam dump, depositing over 1 MW of thermal power. A water cooling system has been devised for extracting this energy while keeping operational temperatures within range. The existing high neutron fluxes in the beam dump during operation produce activation of both coolant and beam stopper, which also suffers from corrosion into the coolant. The presence of radioisotopes in the cooling water leads to a radiological hazard. Water purification systems are located outside the accelerator vault and accumulate activated products during filtration, requiring a specific radiological shield to comply with target dose rates. Also devices containing large volume of activated cooling water, like N-16 decay pipes, require specific radioprotection analysis and design. This work identifies the most relevant radiation sources due to the activated cooling fluid, which may result in doses to workers, and propose radioprotection measures into the design to mitigate their effect. Particle transport and activation calculations have been performed with MCUNED code and analytical models.

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P3-113 **Radioprotection analysis of the Accelerator Facility of IFMIF**

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IFMIF will be a materials irradiation facility, based on two high intensity (125 mA) 40 MeV deuteron accelerators. Deuterons at this energy interact with matter producing significant secondary particles and material activation. This results in a radiological hazard for workers and public, and also a concern to facility dismantlement considering radioactive waste production. This work summarizes the radiological analysis and design of the Accelerator Facility of IFMIF, including the following aspects:

- Considerations about relevant nuclear data.
- Dose rates estimates during operation.
- Requires waiting times before accessing the vault after operation, and expected doses during maintenance periods.
- Activation of air and cooling water.
- Solid waste production. The MCUNED code has been used for transport and ACAB for activation calculations.

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P3-114 **Reproduction of the ITER benchmark on shutdown dose rate calculation with the enhanced R2S-UNED code system**

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An ITER shutdown dose rate (SDR) test case was proposed by ITER IO in September 2010 to compare several SDR tools in a fusion relevant problem with simplified geometry. The benchmark consists in the calculation of the residual gamma dose rate 106 s after shutdown, following an ITER-like irradiation scenario. The irradiated device is a cylindrical assembly featuring dimensions, materials, shielding and streaming characteristics of typical ITER equatorial port. R2S-UNED is a mesh-based R2S (rigorous two steps) system coupling the transport code MCNP and activation code ACAB to perform residual dose rate calculation. The previous version of R2S-UNED incorporated several improvement respects to other R2S systems (MCR2S or R2Smesh) used in the ITER benchmark. Improvements such avoiding the PTRAC run to determine the material volume fractions inside each voxel or considering the emission of decay gammas with spectra corresponding to the real activated material inside voxels enclosing various materials, were already implemented in R2S-UNED.

The enhanced R2S-UNED system, instead of using a voxel averaged neutron flux for the activation calculation, is now able to evaluate separately the neutron flux for each material enclosed in the same voxel. This improvement avoids overestimating the activation of voxels enclosing both material and vacuum – in such cases, due to the neutron streaming component in vacuum, the voxel averaged neutron flux used to activate the material is much higher than the correct one. Thus, the mesh size is no longer limited by geometrical dimensions but only by neutronics considerations. In this paper the results of ITER benchmark exercise performed using enhanced version of the R2S-UNED system are presented and advantages of the new R2S-UNED are compared to other R2S versions.

P3-115 **Assessment of radioactive wastes from a DCLL fusion reactor: disposal in "El Cabril" facility**

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Under the Spanish Breeding Blanket Technology Programme TECNO_FUS a conceptual design of a DCLL (Dual-Coolant Lithium-Lead) blanket-based reactor is being developed. The dually – cooled breeding zone is composed of He/LiPb and SiC as material of the liquid metal flow channel inserts. Structural materials are ferritic-martensitic steel (Eurofer) for the blanket and austenitic steel (SS316LN) for the vacuum vessel (VV) and the cryostat. In this work, the amount of radioactive wastes generated is calculated. Then, they are assessed in order to determine if they can be stored as low and intermediate level radioactive waste (LILW) in the Spanish near surface disposal facility of "El Cabril". Also, unconditional clearance and recycling waste management options are studied.

The neutron transport calculations have been performed with MCNPX code, while the ACAB code is used for calculations of the inventory of activation products and for activation analysis, in terms of waste management ratings for the options considered. Preliminary results show that the cryostat can be disposed in "El Cabril" as an intermediate level radioactive waste. Regarding the VV, only the outer layer can be stored in "El Cabril" as an intermediate level radioactive waste. On the other hand, the eurofer blanket structure and the channel inserts do not fulfill the "El Cabril" disposal requirements in less than 100 years. This is due to the presence of niobium, which is also a troublesome element for the "El Cabril" precluded part of the VV. Finally, it is seen that most of the concrete-made biological shield can be managed through clearance after a cooling period of about three months. In addition, the difficulty of steel materials to satisfy recycling conditions has been discussed.

P3-116 **Safety Analysis of Radioactive Waste Management and Decommissioning of IFMIF Accelerator Facilities.**

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During the operation of IFMIF accelerator, the interaction of radiation with matter can lead to the activation of the accelerator facilities components and surroundings systems. Also as a result of maintenance operation and during decommissioning of the installation, significant amounts of radioactive waste are evacuated and shall be managed according to the radiation protection requirements. Managing of radioactive waste, therefore, involves a variety of technical and operational activities that may extend over a long period of time. The timescale for waste management and decommissioning strongly depends on the adopted approach and techniques, on the type of facility and on the radionuclide inventory.

Present paper summarises the origins of radioactive material in the IFMIF accelerator facilities, focusing on solid materials and liquid and gases (tritium) radioactive wastes for treatment systems. An analysis has been performed of the items with significant levels of activation, to categorise their disposition at various times after the final dismantling and decommissioning of IFMIF. Also, the paper gives an overview of the practices in radioactive waste management and decommissioning of IFMIF accelerator facilities with the objectives to accomplish with the a establish radiation safety requirements in order to minimize the hazardous waste arising from the operation and decommissioning, both in its quantity and level of hazard potential.

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P3-117 **Radiological Impact on Members of the Public due to the Releases from the Accelerator Facility of IFMIF.**

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The International Fusion Materials Irradiation Facility (IFMIF) is designed to be a materials irradiation facility where candidate materials to be used in future fusion designs could be tested. IFMIF is based on two high intensity (125 mA) 40 MeV deuteron accelerators which forms the Accelerator Facility. Interactions of these deuterons with surrounding materials in the acceleration generate activation isotopes which can be released to the environment. These releases could be produced during normal operation of the installation, but also due to external or internal events which break the multiple confinement barriers used as protection of IFMIF. Consequences of releases to the environment in both, normal and accidental conditions should be assessed following international guidance.

This work analyzes the radiological impact to the members of the public due to gaseous or liquid releases produced in both conditions. Dedicated software for transport and dispersion of radioactive releases used included CROM for the case of normal operation discharges and JRODOS for accidental conditions. In both cases conservative assumptions and generic models were applied. In normal conditions likely releases of activated air and leaks from the cooling systems were assumed. For accidental conditions both Base Design Accident (BDA) and Beyond Base Design Accident (BBDA) were defined to obtain consequences covered by design and extreme, very unlikely, events.

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P3-118 **Testing of FW mock-ups under representative operating conditions**

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Research Centre Rez (RCR) has carried out a number of testing and R&D activities in support of the development of the ITER plasma facing components (PFC) and blanket. These contributions have led to the development of numerous experimental facilities, especially BESTH device and irradiated TW3 rig. The BESTH device was developed for a first wall (FW) mock-ups testing. On this device seven FW mock-ups were tested. Five of them were standard 12 000 cycles long thermal fatigue tests of FW mock-up from various suppliers: from China, Russian Federation, South Korea, EU and USA; last two mock-ups were tested for life-time durability, lasting 30 000 cycles. All mock-ups were tested to heat flux of 0.625 MW/m2 in cycles consisting of 30 seconds heating up, 180 seconds of full power, 30 seconds cooling down and 60 seconds of power off. The irradiated rig TW3 allowed an in-pile thermal fatigue testing of actively cooled primary first wall mock-ups in nuclear reactor core was carried out to check the effect of neutron irradiation on the Be/CuCrZr joints under representative first wall operation conditions. Developed rig were located in a core of the LVR-15 experimental nuclear reactor in the RCR. The rig TW3 generated 7 minutes long cycles during the in-pile operation. All mock-ups were tested to heat flux of 0.5 MW/m2 in cycles consisting of 30 seconds to heating up, 180 seconds to keep full power, 30 second to cool down to zero power and 180 seconds to keep on zero power. Neutron fluences and dpa values reached average values 8.4e1020 - 7.6e1020 cm-2 and 0.456 - 0.598 dpa for 17 000 cycles. The paper describes both experimental devices and brings results of the FW mock-up long-time testing in photos, tables and graphs.

Topic H FNT Special Neutron Sources

P3-119 **Molecular dynamics simulations to evaluate the effects of stacking fault energy on interactions between a line dislocation and irradiation-induced defect clusters formed by fusion neutron irradiation**

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Mechanical property changes are a key factor to determine the lifetime of fusion materials. The irradiation hardening is one of the serious issues for the fusion materials. The 14MeV neutrons generated by D-T reaction create irradiation-induced defects, and their interaction with a line dislocation is one of the micro mechanisms for irradiation hardening. In this study, we prepared seven sets of interatomic potentials for FCC metals. These potentials were originally developed from Cu, and SFE were changed from 14.6 mJ/m2 to 186.5 mJ/m2, while the other material parameters were kept almost identical. We conducted molecular dynamic simulations by using these potentials to simulate the interaction with an edge dislocation and a spherical void under applied stress. In low SFE cases, two partial dislocations are depinned separately from the void and there are two stress peaks according to each partial dislocation depinned. In this case, irradiation hardening is relatively low. On the other hand, in high SFE cases, two partial dislocations are depinned almost simultaneously from the void, where we can observe higher irradiation hardening. The interaction morphology depends on both SFE and void size, which correspondingly changes irradiation hardening. In the lowest SFE, the former interaction is observed even with the relatively lager voids.

This study is for the first time to evaluate the interaction with a line dislocation and a void by setting the SFE equivalent to austenitic stainless steels. These findings are essential to build up a model which can predict degradation of FCC metals under fusion neutron irradiation.

P3-120 **A new method and experiment for real-time three-dimensional displacement measurement of EAST magnets based on computer vision**

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The Experimental Advanced Superconducting Tokamak (EAST) is one of the running experimental facilities to demonstrate the scientific and technical feasibility of fusion power. The superconducting magnets provide the magnetic fields necessary to control the position and shape of the plasma. The magnets are installed at room temperature and work at temperature around 4K. During the cooling process from normal to cryogenic temperature, magnets will shrink; this will change the magnets position. So acquiring magnets position status in real-time is of great significance in controlling the plasma and improving the research and design of the device. At present, measuring magnets displacement adopts the method of using line resistance displacement sensor, the disadvantages of this method are inexact measurement results and hard to process the data. In order to acquire magnets moving data more accurately and effectively, a modified three-dimensional measurement model is built, computer vision method is adopted to process the data, and real-time measurement results can be obtained using the edge detection algorithm. This method can eliminate the influence of various factors on the test. Now the new measurement system has been installed inside the experimental hall in 2010, the measurement system has been tested in the entire process of EAST experiment in 2011, after the annual experiment, the measurement system was upgraded and reformed. The measurement system stably and reliably obtains a large amount of experimental data in 2012; the data of the movement from room temperature to around 4K and the shrinkage at extreme low temperature have already been got.

P3-122 **Optimization of MGI in JET using TOKES code**

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Disruption is a spontaneous break down of the tokamak discharge stability, due to inherent processes in plasma. Unmitigated disruption in ITER will damage the first wall causing its melting, brittle destruction and erosion. Mitigation of the first wall damage is one of the most important issues for ITER performance. Experiments performed in modern tokamaks have proved that fast injection of a massive amount of noble gas (MGI – massive gas injection) can effectively mitigate the disruption, transforming both, the plasma energy and the poloidal magnetic field energy into radiation, which loads the first wall more uniformly than the unmitigated plasma impact.

The TOKES code, developed in FZK – KIT for integrated simulations of the plasma performance and plasma–wall interactions in tokamaks is used for simulations of MGI. Previous simulations of the plasma evolution and its thermal energy irradiation during thermal quench of MGI in JET have been done for verifications of the physical models implemented in TOKES. Verification of the TOKES simulations is essential for predictions of the first wall damage in ITER. MGI is an emergency method for mitigation of disruptions. A side effect of MGI is contamination if the vacuum vessel with large amount of noble gas, which should be pumped out for further operation of tokamak. Hence, it is important to determine the minimum amount of the noble gas for mitigation of the disruption.

In this paper the experimental data on electron density during MGI available from JET is compared with the results of TOKES simulations with the aim to verify the TOKES results and additional simulations has been performed to determine the minimum amount of the injected gas for reliable shut down of the discharge.

P3-123 **Development of Self-Powered Neutron Detectors for Neutron Flux Monitoring in HCLL and HCPB ITER-TBM**

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The development of tritium breeding blankets is one of the main challenges on the path of a DEMO reactor. ITER will provide the first experimental data on tritium production performance by testing mock-ups of DEMO blankets, called Test Blanket Modules (TBMs).

The development and testing of nuclear instrumentation for on-line measurement of neutron and gamma fluxes and tritium production is of paramount importance for testing of TBMs in ITER. This instrumentation must withstand the harsh working conditions expected inside the TBM (temperature > 400°C, magnetic field and high level of radiation fluxes). Several detectors have been proposed so far, however no detectors are presently available to withstand the TBM working conditions. A dedicated development program is thus necessary. As part of the European Consortium on Nuclear Data and within a F4E Grant, ENEA and KIT have jointly studied the possibility to develop and test detectors suitable to operate in ITER-TBM. Among the various detectors considered, Self-Powered Neutron Detectors (SPND) seems very promising. SPND produce a current proportional to neutron activation (beta emission) induced in a metal wire called emitter. SPND were developed for nuclear reactors, the commercial SPND detectors are able to fulfil some TBM requests but are not suitable to operate in the hard neutron spectrum available in TBM (low beta activation).

A study was performed to apply SPND to the working conditions expected in the ITER-TBM (both HCLL and HCPB). This paper reports the preliminary results of experimental tests of three commercial SPND. Starting from these results a study of possible new emitter materials has been conducted using the FISPACT code. As a result of these calculations applying typical TBM neutron flux spectra a few materials were found which might be suitable for being used in a SPND detector operating in the ITER-TBM. The realization and test of new SPND suitable for operation in TBM is proposed.

P3-124 **Effect of high energy electron component on detached plasma during ELM-Like plasma**

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The research on dynamic behavior of plasma during pulse plasma flow is a topic in nuclear fusion. In particular, the pulse plasma flow, such as the ELM burst, leads to periodic particle and heat loads in a cycle on the divertor plate. During ELM burst, the electron energy distribution function strongly deviates from Maxwell distribution. Thus, it is difficult to stationary detached recombining plasma in divertor. On the other hand, the ability to produce plasmas with high frequency and small ELMs such as type-II ELM and grassy ELM is an important step towards extending the lifetime of the divertor target plates. We have carried out the experimental observation and modeling of time-dependence of Balmer series emission intensities in hydrogen recombination plasma in a linear plasma device, TPD-Sheet IV. The pulse plasma flow was generated by the switching circuit controlled the electric potential of the next floating electrode of the anode in plasma source. The duration of the pulse plasma was 0.3ms and the frequency of cycle of the plasma was 50-1500Hz. The time-dependence of electron density ne, electron temperature Te, electron energy distribution function fe(E) were measured using Langmuir probe. The ionization and recombination events are discussed by Collisional-Radiative model, taking into account of high energy electrons.

P3-125 **Theoretical Model for the Determination of The Reference Sieverts' Constant and Diffusivity Values for Hydrogen Isotopes in Eutectic PbLi**

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Hydrogen isotopes transport parameters of Sieverts' constant and diffusivity in the eutectic lead lithium alloy are key transport parameters determining tritium management strategies at liquid-metal breeding blanket systems [Helium Cooled Lithium Lead (HCLL), or Dual-Coolant Lead-lithium (DCLL)] as reference blanket options.

Tritium transport parameters of solubility and diffusivity in the alloy will determine the magnitude and kinetics of the induced diffusive tritium flux, mainly from the breeding region in the blanket to the helium cooling loop. In addition, the design (sizing and efficiency) of future tritium extraction systems of the breeding alloy or the He coolant purification system will be defined on the basis of these transport parameters.

In this work a theoretical model which describes the interaction between isotopes of hydrogen and eutectic PbLi sample is developed in case of the particular boundary conditions of the Absorption-Desorption facility available at UPV/EHU. Both experimental stages of absorption and desorption are entirely simulated in order to reduce the present dispersion band in the available experimental data (two orders of magnitude) obtained by different research groups using different experimental techniques.

P3-126 **Thermoluminescence measurements of neutron streaming through JET Torus Hall ducts**

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Neutron streaming through penetrations of ITER structural/shielding materials need to be calculated for the safety assessment of biological shields. Particularly, evaluation of neutron streaming through ducts is a major safety task involving radiation transport computations along long paths and in complex geometries. Therefore, experiments at JET aiming at validating such calculations are of utmost importance, since they would provide unique experimental data in fusion environment and enable validation of the safety calculations made for ITER.

For this purpose, thermoluminescence detectors (TLD) were used for dose measurements at JET. Several hundreds of LiF detectors of various types, standard LiF:Mg,Ti and highly sensitive LiF:Mg,Cu,P were produced. LiF detectors consisting of natural lithium are sensitive to slow neutrons, their response to neutrons being enhanced by 6Li-enriched lithium or suppressed by using lithium consisting entirely of 7Li. Pairs 6LiF/7LiF detectors allow distinguishing between neutron/non-neutron components of radiation field. For detection of neutrons of higher energy there is a need of moderators. Cylindrical moderators (25 cm diameter and 25 cm height) have been produced from polyethylene (PE-300) rods. All TLDs, located in the centre of cylindrical moderators, were installed at eleven positions in the JET hall and the hall labyrinth in July 2012, and exposure took place during the last two week of experimental campaign. Measurements of the gamma dose and of the neutron fluence were obtained for all positions over a range of about five orders of magnitude variation. The experimental results are compared with calculations using MCNP code. The results confirm that the TLD technology can be usefully applied to measurements of neutron streaming through JET Torus Hall ducts. New detector positions, further in the labyrinth and ducts, will be investigated in the next measurement campaign. Detectors positioning inside moderators will be improved to reduce the shadowing effect observed for detectors containing 6Li.

P3-127 **Technical aspects and manufacturing methods for JT-60SA toroidal field coil casings**

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In the frame of the Broader Approach program ENEA, under the F4E general coordination, is responsible of the supply of eighteen Toroidal Field (TF) coil casings for the JT-60-SA tokamak,. According to the technical specifications issued by ENEA and the JT-60SA European home team, a contract between ENEA and Walter Tosto started on July 20012 for the construction of two different sets of 9 TF coil casings, to be progressively supplied from 2013 to the end of 2015. Each TF coil casing consists of four main components: one "Straight Leg Outboard" and one "Curved Leg Outboard' both with their own cover, "Straight Leg Inboard" and "Curved Leg Inboard". The casing components are segmented in forgings and plates made of FM316L-NL. The straight leg outboard is composed by two wings welded to a central core and two elbows welded at the ends with a cooling channel installed inside. Elbows of straight leg outboard are segmented in two half-elbows machined from 1 rough forging and welded to the central core made by plate. Welding of wings to central core will be performed in EBW (Electron Beam Welding) and the straight part will be welded to the elbows by NGTIG (Tig Narrow Gap) process. The curved leg outboard is composed by two wings welded to a central core for a final shape of "D". Other supports will be welded by TIG or Electrode process. This paper describes the technical design solutions, the manufacturing methods defined and the particular processes adopted, such as welding (EB, TIG), non-destructive examinations (NDE), vibration stress relief (VSR), laser tracker survey, most of which have been validated by the construction of two different sets of full scale mock-ups representing the straight and the curved legs.

P3-129 **Stability of the LIPAc beam dump to vibrations induced by the cooling flow**

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The beam dump of the LIPAc accelerator, designed at CIEMAT, should be able to dissipate the great amount of heat released by the 125 mA 9 MeV deuteron beam. It consists of a copper cone, 5 mm thick, 250 cm long and 30 cm base diameter, placed with its base facing to the beam and coaxial with it. The heat is removed by a pumped water cooling loop and a heat exchanger. The flow rate, variable up to 120 m3/h, as well as the geometry of the cooling channel have been carefully defined to assure an optimum cooling, keeping the temperature and pressure of the water below its boiling point.

The high water velocity in the cooling channel (between 4 and 8 m/s) and the turbulent regimes involved may compromise the mechanical stability of the slender beam dump structure and cause damages due to flow induced vibrations. Since the system is too complex to be studied theoretically, some tests have been carried out to evaluate its behaviour in normal operating conditions. These tests, performed on a model built at 1:1 scale and replicating exactly the final version, have been focussed on finding experimentally the main vibration modes and the responses to the different working flow rates. This modal analysis, together with the results of the measurements of the vibration characteristics obtained at several positions of the cone, is presented here. With amplitudes not greater than 500 mg rms in any case, the structure has proved its practical immunity to flow induced vibrations, thus validating its design and construction methodology.

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P3-131 **The high flux plasma generator MAGNUM-PSI**

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In Magnum-PSI (MAgnetized plasma Generator and NUMerical modeling for Plasma Surface Interactions), a quasi steady state axial magnetic field up to 1.9 T is generated to confine a high density, low temperature plasma of a wall stabilized dc cascaded arc to a magnetized plasma beam. It aims at conditions that enable fundamental studies of plasma-surface interactions in the regime relevant for fusion reactors such as ITER: 10^23 - 10^25 m-2 s-1 hydrogen plasma flux densities at 1-5 eV [1]. At these low temperatures, residual neutrals may compromise the controllability over the plasma conditions. Therefore, a differentially pumped vacuum system has been designed on the basis of neutral gas simulations that keeps the neutral gas flow to the target region sufficiently low, while the plasma source puts in large quantities of neutral gas.

In this contribution, experiments will be presented that characterize the performance of Magnum-PSI. Magnetized hydrogen and deuterium plasmas have been produced with electron densities up to 6.5×10^20 m-3 and electron temperatures up to 3.7 eV. Target calorimetric data in combination with pressure measurements demonstrate that the flow to the target region is determined by the incoming ion flux and not by neutrals produced directly by the plasma source [2].

The experiment will be upgraded by a superconducting magnet so that the fully commissioned Magnum-PSI machine will continue operation as a steady state device, expanding the operational space to high fluence capabilities.

[1] G. De Temmerman et al, 27th symposium on fusion technology (2012)

[2] H.J.N. van Eck et al, Appl. Phys. Lett. 101, 224107 (2012)

P3-132 **Study of the Use oF ESS-BILBAO for Irradiation of Nuclear Fusion Materials**

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Nuclear materials must withstand severe irradiation conditions that determine their performance. In particular, these materials must tolerate high neutron doses at the end of their lifetime. A common problem when dealing with nuclear materials is the lack of neutron irradiation facilities appropriate for materials characterization and qualification. For the study of nuclear fusion materials one needs high fluxes of 14 MeV neutrons. Several facilities are nowadays available, however, none of them completely fulfills the fusion requirements. Some installations are based on fission reactions with neutron energy ~1 MeV, whereas others are based on accelerator driven sources with energies from MeV to GeV.

In this work two accelerator driven neutron sources, ESS-Bilbao and ESS-Lund are considered for fusion damage studies. We will compare the neutron spectra that can be obtained with the expected ones in fusion power plants, in particular, in plants projected within the framework of the European laser fusion project HiPER.

The resulting neutron damage materials is analyzed for, a) iron as a reference structural material and b) silica as an optical material. For the analysis several features have been studied such as neutron fluxes, doses, PKA spectra, gas production and dpa estimation. The analysis is performed with MCNPX, NJOY and self-developed codes.

The results show that damage rates in ESS-Bilbao are low compared with demonstration or commercial fusion plants. However ESS-Bilbao turns out very interesting to study phenomena that appear at low dose rates, such as damage evolution in optical components. On the other hand, ESS-Lund will produce too energetic neutron (and proton) spectra to mimic realistic fusion irradiation conditions.

The main conclusion is that the small accelerator driven neutron source (ESS-Bilbao) will be useful for the analysis of optical components (and probably other ceramic-based materials). Two are its major advantages: neutron spectra that match well the fusion spectra and versatility to perform a wide range of experiments. The main drawback is the impossibility to attain high damage rates and therefore carry out relevant studies of structural materials. Other applications of ESS-Bilbao as a neutron irradiation facility are feasible in addition to the application for fusion materials discussed in this work.

P3-133 **Calculation of the changes in neutron detector response due to the gradual upgrade of the JET torus**

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The Joint European Torus (JET) entered a shutdown state in 2009, during which the carbon plasma-facing wall was replaced with the 'ITER-like wall'; a combination of beryllium, tungsten and CFC. This was the most recent major change of the torus configuration, which has changed several times in the past. The paper presents calculations of the major contributors to the change in the diagnostics response for several past stages of the JET torus.

The last calibration of detector systems with a 252Cf neutron source was at JET performed more than twenty years ago and it is only for that particular state of the torus, that a direct comparison of measurements and transport calculations can be performed. Each upgrade of the torus, however, affects the neutron field and hence the calibration factors for individual neutron detector systems, which are the most important indicators of fusion power. Calculations are therefore important for an understanding of the detector response.

The emphasis is on the most important in-vessel neutron diagnostics – the activation system and in particular the KN2-3U irradiation location. The calculations of the neutron and gamma flux distributions inside the JET torus are performed with the Monte Carlo method (using the MCNP code).

Response values, calculated for several stages in the torus development are compared, starting with the torus in the state as during the last calibration in the 1980ies. The most important single change in the response was found to be due to the vertical upward shift of the plasma center due to the introduction of the divertor and the introduction of mushroom limiters close to the KN2-3U. The exchange of the carbon wall with the ITER like wall resulted in a slight increase of the flux inside the vessel and consequently of the KN2-3U response.

Topic I Repair and Maintenance

P3-135 Preliminary Assessment for Dust Contamination of ITER In-Vessel Transporter

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After plasma operation of ITER, radioactive dust will be accumulated in the vacuum vessel (VV). The In-Vessel Transporter (IVT) will be installed in the VV to remove and install the shield blanket modules for maintenance or exchange. IVT itself also needs to be maintained regularly in the Hot Cell Facility (HCF). It should be considered that maintenance workers are exposed to the radioactive dust adhering to the IVT surface. In this study, the dust contamination on IVT is evaluated to assess the level of exposure during maintenance work in HCF.

In ITER, there are several scenarios to be considered for evaluating the dust contamination and the most conservative one assumes that all dust in VV will adhere to the surface of VV and IVT. This corresponds to the safety-side assessment. Based on this scenario, the effective dose rate from radioactive dust has been estimated. This study concentrates on external exposure only and the effect of tritium is left for further study. As a result, it has been found that W-181 and Ta-182 are the dominant nuclides for the effective dose rate. Assuming that all dust is W-181 or Ta-182, the effective dose rate reaches about 400 μSv/h and 100μSv/h, respectively. On the other hand, applying the dose limit determined by the ITER project and considering the estimated maximum maintenance time, the effective dose rate should be limited at around 4μSv/h. Decontamination processes regarding IVT, such as gear and linear guide mechanism, have been investigated and the dose rate after the decontamination processes has been evaluated. Even though dust is removed from the surfaces where decontamination is supposed to be possible, the dose rate exceeded the dose rate limit. To satisfy the dose rate limit, decontamination and maintenance plan including in decontamination of dust, radiation shield system and working time reduction, is proposed.

P3-136 **Erosion evaluation capability of the IVVS for ITER applications**

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The IVVS (In Vessel Viewing System) is a 3D laser scanner based on AM (Amplitude-Modulated) ranging technique able to obtain high-resolution intensity and range images in hostile environments. The system was designed to be compliant with ITER vessel requirements (high temperature, gamma ray, neutron flux, high magnetic field and ultra-high vacuum). In the framework of a Fusion for Energy grant, an investigation of the IVVS metrology performances was required to evaluate the device capability to estimate erosions on ITER first wall and divertor. In ENEA Frascati laboratories, a method and a computational procedure were developed and applied to an eroded target simulating the ITER vessel components. The target is an aluminum alloy plate containing geometrical markers and four eroded areas with different depths (2.1mm, 0.5mm, 0.3mm, 0.1mm). During target realization, the plate weight was measured by a precision balance before and after the erosion process. The final purpose of the IVVS experiments was the evaluation of the amount of eroded material in the target under investigation and consequently the estimation of the missed weight. Experimental tests were carried out positioning the target at a distance of 4m from the IVVS probe and scanning it with several incidence angles. High-resolution (sub-millimetric) 3D models of the investigated target were acquired and then post-processed to obtain an estimation of the eroded volume and weight. The weight of the eroded areas, evaluated by IVVS and measured by the precision balance, was finally compared under different experimental conditions, so the errors on weight losses obtained using the IVVS and the computational procedure were estimated. Additional test campaigns are planned after the implementation of upgrades of IVVS system with the aim to obtain a complete characterization of the IVVS capabilities. The main results obtained during laboratory tests and data processing will be presented and discussed.

P3-137 **Remote Handling Operator Training at JET**

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Remote handling operation has been carried out on the Joint European Torus for over 15 years with in excess of over 20,000 hours of in-vessel remote handling activities. During this period the operations team has developed to provide the unique skills required to achieve the demanding remote operational campaigns undertaken today. This paper describes the challenges experienced in recruiting, training and retaining the remote handling system operators and examines the evolution of the team profiles currently used. The operations team have a number of key skills that vary depending on the role of the team member. A degree of overlap in skills is essential to ensure that operations can continue with the temporary loss of an operator. The remote handling system at JET utilises two Articulated Booms 12.5 and 9.4 metres in length. The 12.5 metre boom is fitted with a force reflecting servo manipulator (MASCOT) and the 9.4 metre boom is used to transport tooling and components to and from the Torus. MASCOT is a two armed master slave system; each arm has 7 degrees of freedom with the capability to carry a mass of up to 20kg. The three main operator requirements are: MASCOT Operators, Articulated Boom operators and operations engineers. This involves training in a number of unique areas including virtual reality, operations documentation system (ODS) and a number of different man machine interfaces (MMIs). Due to the unique skills required, recruitment of the primary MASCOT operators has by far been the most challenging of these roles to provide. The content of this paper documents the development of the JET remote operations team over the past 15 years, and should provide a resource of value for future tokamak remote handling operations such as ITER and DEMO.

This work, part-funded by the European Communities under the contract of Association between EURATOM/CCFE was carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission. This work was also part-funded by the RCUK Energy Programme under grant EP/I501045.

P3-138 **The amalgamation of imaging with virtual reality to optimise design and maintenance within a tokamak environment**

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The development of JET has seen many tools being used to optimize the design and maintenance processes in order to maximize machine availability for experimental operations and minimise the duration of planned interventions. It is now commonplace to have multiple globally based design centres working with common design data. This is uncovering many challenges that need to be addressed to ensure design excellence and quality assurance are maintained. This has proved to be especially key in remote handling activities. The effective sharing of accurate design data, both existing and proposed and including analytical and graphical formats, is one of these challenges. This paper highlights the combination of two additional aids to supplement the CAD data that can provide a platform for improved design evaluation. Following the EP2 shutdown, there was an identified need at JET to maintain and update a historical high quality pictorial record of the torus status at key milestones. Specific, matched sets of images were captured before and after an experimental campaign to provide comparative data for component erosion or damage.

This survey was undertaken using a matched pair of Nikon cameras at pre-defined referenced locations with an option for the creation of stereo images. The second aid, was to use'3DS Max' Software to construct Virtual Reality (VR) environments with 'Xref' files linked to CAD reference data. This allows quick updating of a VR environment when new design data is presented. Any Remote Handling driven system together with any supported component may be assigned specific controlled motion and degrees of movement. This combination of accurate 3D stereo images and motion controlled elements in a Virtual Reality environment can then be used to determine if a task can be realistically undertaken in a defined working volume.

The amalgamation of the pictorial images and VR data to an absolute reference, that may viewed over the web, may also provide a medium that is detailed enough to be used to demonstrate a task without the need for extensive supporting text descriptions.

P3-139 **Progress in Standardization for ITER Remote Handling Control System**

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The ITER Remote Handling (RH) System is made up of 7 major equipment system procurements which are spread across 3 ITER parties:- the ITER Organization (IO), the European Domestic Agency (F4E), and the Japanese Domestic Agency (JAEA). Each procurement is designed to deliver a complete system (operator interfaces, equipment controllers, and equipment devices), and so standardization is extremely important to ensure that the systems can be integrated at the IO to form a homogenous RH System that can be efficiently operated and maintained by the ITER RH team. An integrated control system architecture has previously been defined for the RH equipment systems [1], and work has been continuing to develop and validate standards for this architecture. Evaluations of standard parts and a standard control room work-cell have contributed to an update of the RH Control System Design Handbook, while R&D activities have been carried out to validate concepts for standard solutions to ITER RH problems:- the use of a standard master arm with different slave arms, the achievement of high accuracy tracking of RH operations within virtual reality, and condition monitoring of RH equipment systems.

The standardization efforts have been consolidated through the development of a freely distributable software platform to support the adoption of the ITER RH standards. The RH Core System installs on top of the CODAC Core System and provides the basic platform for the development of ITER RH equipment controller applications. The standardization work has continued in the areas of RH viewing, network communication protocols, and a structured language for programming ITER RH operations. Prototyping has been done on high-level control system applications, and R&D has been carried out in the area of synthetic viewing for ITER RH. These developments will be reflected in a new version of the RH Core system to be produced during 2013.

P3-140 **Progress in the design and R&D of the ITER In-Vessel Viewing and Metrology System (IVVS)**

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The In-Vessel Viewing and Metrology System (IVVS) is a fundamental tool for the ITER machine operations, aiming at performing inspections as well as providing information related to the erosion of in-vessel components, which in turn is related to the amount of mobilised dust present in the Vacuum Vessel. Periodically or on request, the IVVS scanning probes will be deployed into the Vacuum Vessel from their storage positions (still within the ITER primary confinement) in order to acquire both visual and metrological data on plasma facing components (blanket, divertor, heating/diagnostic plugs, test blanket modules).

Recent design changes of the six IVVS port extensions located at the divertor level implied the need for a substantial redesign of the IVVS plug. This on-going design work, complemented by experimental activities, aims to bring the integrated IVVS concept – including the scanning probe and its deployment system – to the level of maturity suitable for the Conceptual Design Review.

This paper gives an overview of the various design and R&D activities in progress: plug design integration, probe concept validation under ITER-like environmental conditions, development of a metrology strategy, gamma, neutronics and mechanical analyses, gamma and neutron irradiation tests.

P3-141 **Development of remote pipe cutting tool for divertor cassettes in JT-60SA**

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Remote handling (RH) system is necessary for the maintenance and repair of in-vessel components of JT-60SA. JT-60SA equips lower divertor cassettes. Cooling pipes, which connects the divertor cassette and the vacuum vessel with bellows in the outboard side, are required to be cut and welded in the vacuum vessel by RH system. The space for RH system is very limited inside the vacuum vessel, especially in the divertor cassette. Thus, the cooling pipes are required to be cut from inside. The outer diameter, thickness and material of the cooling pipe are 59.7 mm, 2.7 mm and SUS316L, respectively. Remote pipe cutting tool for JT-60SA divertor cassette has been newly developed. Cutting tool head equips a disk-shaped cutter blade and four rollers which are subjected to the reaction force. The cooling pipe is cut by rotating the tool head with pushing out the cutter blade. Newly developed cutting tool indicates that the cooling pipe is cut by pushing out the disk cutter blade up to 30.5 mm in radius, i.e. 61 mm in diameter. The divertor cassette is repaired outside the vacuum vessel after carrying the divertor cassette out. In the case of replacing the inner target plate, the cooling pipes connecting the inner target and the divertor cassette are required to be cut from the inside. However, the pipes have no bellows unlike the cooling pipe between the divertor cassette and the vacuum vessel. When the pipe cutting tool cuts the pipe, the blade pushes the cooling pipes apart. Thus two cooling pipes, one is inlet pipe and the other is outlet pipe in a divertor cassette, are required to be cut at the same time. The newly developed pipe cutting tool is also able to cut these two pipes at the same time.

P3-142 **Using a Data Centric Event-Driven Architecture approach in the integration of real-time systems at DTP2**

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Integration of heterogeneous and distributed systems is a challenging task, because they might be running on different platforms and written with different implementation languages by various organizations. Service-oriented architecture (SOA) is an architectural design pattern that has been developed to solve this problem in enterprise information systems, typically using Web services and XML based protocols. This approach has some downsides when applied to deterministic systems, e.g. uncertain communication latencies due to time coupling between services, increased development complexity and a possible single-point-of-failure caused by a central message broker. A better fit for deterministic systems is a Data Centric Event-Driven Architecture (DC-EDA) that focuses on providing a global data space instead of services, and decouples the services with respect to time.

This paper focuses on the implementation of inter-subsystem communication in a prototype distributed remote handling control system developed at Divertor Test Platform 2 (DTP2). The control system consists of a variety of subsystems, such as the Operations Management System (OMS) and the hard real-time Equipment Controllers (EC). These subsystems are responsible for specific tasks and have been developed on various platforms and programming languages. Adaption of a DC-EDA based communication architecture can benefit the system by making it more flexible, uniform and robust.

The implementation presented in this paper uses the Object Management Group's (OMG) standard specification for Data Distribution Service (DDS) for ensuring communications interoperability. DDS lacks a centralized broker, thereby avoiding a single-point-of-failure. One of the greatest challenges in integrating a system with a DC-EDA based approach is in defining the global data space model. A model that fits the current DTP2 system is presented, along with comments on application to larger systems. The performance of point-to-point communication is also evaluated with and without the presence of additional lower priority network load to ensure applicability to real-time systems.

P3-143 **An architecture and a demonstrator for the ITER Remote Handling Viewing System**

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The Remote Handling of ITER relies on the principle of 'man-in-the-loop' remote operation. The viewing system is tasked to providing the operators with the visual feedback. The ability to perform efficient and safe man-in-the-loop remote operations is directly linked to the viewing capabilities. Functionally, the main issues are: Image switching: selecting one or more images from a specific camera and displaying it on one or more specific monitors in the RH Control Room Image control: access all the 'features' and 'attributes' related to the image/video stream capturing such as pan, tilt, zoom, focus, contrast, brightness, colour balance, image size, frames per second, etc. Image improvement: enhancement of the image content such as de-noising, edge sharpening, contrast stretching, etc. The complexity of ITER RH is unprecedented and bear little resemblance to the one of JET, designed some 30 years ago. This is reflected also in the RH Viewing System, that is expected will require anything between 500 and 800 cameras. The massive data amount produced will use TCP based connectivity both for data transport and for switching, and other important issues such as video stream compression/decompression, front end processing, and latency will need to be addressed. These solutions can leverage on the rapid progress of industrial image processing in the last decades and adopt state-of-the-art solutions such as the latest digital camera sensors and interconnectivity standards, distributed front-end processing, multicore CPU and FPGA-based porting of advanced compression and enhancement algorithms. Scalability and modularity of the implementation will also be an important design feature, taking in due consideration the time scale of the project and the expected technology changes specific to the machine vision. The paper will illustrate the main needs and criteria that have led to today's architecture, as well as reporting the early results of the ITER Viewing Demonstrator (IVD).

P3-144 **Progress in the Conceptual Design of the ITER Cask and Plug Remote Handling System**

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ITER maintenance requires transportation of in-vessel components and Remote Handling Systems (RHS) to and from the Vacuum Vessel, at all levels of the Tokamak building, to the docking stations in the Hot Cell building via dedicated galleries and lift. The Cask and Plug Remote Handling System (CPRHS), a.k.a Transfer Cask System has been adopted as the solution to provide nuclear confinement and avoid spread of contamination during these operations.

The cask geometry and in-cask handling equipment vary according to the components or systems being transported. Four different types of cask are required related to maintenance of in-vessel port plugs or systems, namely; Torus cryopump, Upper & Equatorial Port Plugs and the In-vessel Viewing System (IVVS). In addition two cask typologies are required to transport other RHS which perform maintenance operations within the Tokamak, namely; Divertor RHS and the Blanket RHS a.k.a. In-Vessel Transporter (IVT). Also, due to nuclear safety requirements it will be necessary to provide dedicated systems to endure remote rescue of a failed cask. Following completion of the conceptual design by the ITER Organisation (IO) Fusion for Energy will take responsibility for the CPRHS through to delivery of the circa twenty casks which will make up the complete fleet.

Fusion for Energy has been working with the IO, under an ITER Task Agreement, in the development of the conceptual design with the aim of ensuring the feasibility of the cask operations and enabling the technical requirements to be clearly defined. This paper will discuss the development of the conceptual design to date under this agreement and present the main technical risks associated with the finalisation of the design and subsequent manufacture of the CPRHS units. Due to the nuclear confinement function the CPRHS is defined as safety critical system and as such will need to be design, manufactured, tested and commissioned with respect to compliance to the French Autorité de sûreté nucléaire (ASN) nuclear safety requirements related to ITER. This will bring significant challenges to the CPRHS development which will also be discussed in this paper.

P3-145 **Robot Vision System R&D for ITER Blanket Remote Handling**

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For maintenance of ITER, a system to remotely handle the shield blanket modules is necessary because of high gamma-ray field. It is called the ITER blanket remote handling system (BRHS). Blanket handling will be carried out by robotic devices such as power manipulators. To avoid contact with the surface of modules, the manipulator should have a non-contact sensing system to install and grasp a module. The manipulator is required to be accurate within 5 mm in translational motion and 1 degree in rotational motion.

The Robot Vision System (RV) was adopted as the non-contact sensing system. RV is a sensing method using cameras. Additionally, to satisfy the requirements, three widely used methods of RV were adopted: Stereo Vision, Visual Feedback and Visual Servoing. Stereo Vision is a RV method using two cameras. In Visual Feedback, the manipulator moves to the target position in many sequential steps. In Visual Servoing, the manipulator moves in order to fit the current picture with the target picture. Also, note that it is completely dark in the vacuum vessel and lighting is needed.

Tests for grasping a module using those three methods were carried out and the measuring error of the RV system was studied. The resolution of the used cameras is consistent with a video camera tube, which is a radiation tolerant camera. These tests simulated the ITER vacuum vessel included a lighting environment: extrinsic light was shut out by a curtain and lights were placed on the end effector of the manipulator and places corresponding to the ports in ITER.

The results of these tests were that the accuracy of the manipulator's movements was within 1 mm and 0.3 degrees using RV. This satisfies the requirements; therefore, it is concluded that RV is suitable as the non-contact sensing system for the ITER BRHS.

P3-146 **Shutdown dose rates in a DEMO remote handling environment**

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In Europe the work on the specification and design of a Demonstration Power Plant (DEMO) is being coordinated by EFDA in the Power Plant Physics and Technology (PPP&T) department. Within this framework an important activity package is devoted to the assessment of candidate remote maintenance schemes and solutions. The characterization of the shutdown radiation environment is an important input in the feasibility assessment and design of remote handling (RH) equipment. This paper addresses shutdown radiation field analyses conducted in the frame of the EFDA 2012 work programme on the basis of a provisional HCLL DEMO1 model. The aim is to predict the shutdown dose rates at relevant cooling times in key locations where RH tools are expected to operate. Dose rates at in-vessel locations and at ports exits have been calculated for a considerable number of materials used in RH equipment.

In computing the dose rate mapping the Rigorous two-step (R2S) code coupling methodology implemented in the UNED and KIT systems are used to perform activation calculations on meshes superimposed to the real geometry and to determine the gamma dose field resulting from neutron activation. Activation of all components of the reactor geometry has been considered in this study. In-vessel absorbed dose rates are found to range from around 2kGy/h at 1 week following shutdown to 0.5kGy/h at 1 year following shutdown. In the ports, the absorbed dose rate levels range from 90 Gy/h in the lower port to 5 Gy/h in the upper port at one week following shutdown. At one year following shutdown, the dose rates decrease to 15 Gy/h and 0.7 Gy/hr respectively.

Additional contributions, such as the biological dose rates at the port openings and the impact of an increased irradiation time to scale results from DEMO to a commercial power plant have also been addressed.

P3-147 **An Optimization Method of Maintenance Scenarios for Nuclear Facility**

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During the maintenance of activated component of nuclear facility, such as ITER, workers could be badly injured by radiation emitted by the activated component. So, it is very important to evaluate the dose of workers in the reactor building and prevent them from absorbing too much exposed dose according to ALARA principle, i.e. to reduce the dose as low as reasonably achievable.

An optimization method has been developed to optimize the maintenance scenarios of nuclear facility, and it has been integrated into RVIS, which is a visualized program for nuclear and radiation safety simulation. The process of the method is as follows: 1) Maintenance trajectories defining in a visualized scene, which plots dose field overlaid on geometry, according to maintenance plan. ALARA analyst can intuitively pick the location which the avatar can walk or stand in to avoid the high dose rate region. 2) Virtual roaming simulation based on pre-defined trajectories and organic dose assessment by using Rad-HUMAN, which is a whole-body computational phantom of Chinese adult female developed by FDS Team,.3) Visualized analysis of multi-scenarios comparison result and auto optimizing of maintenance scenarios based on Multi-Object Optimization algorithm.

In order to verify the rationality and validity of the method, the maintenance process of activated component of ITER has been selected as a test. The test result showed that the radiation dose of organs and human body of every scenario could be analyzed straightly and the optimized maintenance scenario has strictly followed the ALARA principle and effectively reduced the occupational exposure dose. Experiments have proved that the maintenance scenarios optimization method could be applied not only to the maintenance scenarios' optimization of ITER, but also to other fusion and fission nuclear systems.

Keywords: Activated component, ALARA principle, Maintenance scenarios optimization, Organic dose assessment

P3-148 **The Software Protocol Design: Communication and Control in Multi-task Robot Machine for ITER Vacuum Vessel Assembly and Maintenance**

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In the ITER vacuum vessel (VV) manufacturing process, the VV consists of nine individual sectors which would be jointed seamlessly by the welding process. Such a manufacturing task involves various machining process, which includes the procedures of gap profile scanning between the adjacent vacuum sectors, tungsten stuffing-block handling, adjacent vacuum sector welding, welded surface milling, defective welded area cutting in assembly process as well as the flexible housing boring, threading and milling in maintenance process.

Currently none of the commercial robot machine is available to deal with all these tasks from inside of the vacuum vessel due to their big size and high weight. A small 10 degrees of freedom (DOF) mobile robot machine for the specific machining purposes has been developed in Lappeenranta university of technology.

Since all these capabilities are integrated in one robot machine, there are abundant and complicated communications between the software graphic user interfaces (GUIs) and the bottom control system. For the reason that different machining capability has the different command logics, different data sets as well as the different control strategies, thus a specific protocol is stringently required to guarantee the smooth and real-time data steaming inside the software. In this paper, a particular scalable protocol has been designed to manage the data streaming. According to the data property, such as data for trajectory, data for command or data for status, the protocol is realized by three specific forms: trajectory protocol, command protocol and status protocol. However, each specific protocol consists of three different layers sequentially in a longitudinal direction.

The first layer would provide the template for the general information of all the machining tasks such as the task entries and the priority etc.

The second layer would envelop all the executable data content in which the individual task data are arranged in a latitudinal direction. This arrangement also enables the protocol to be scalable dynamically.

The third layer would provide the communication validation which is used for the transported data completion checking.

In addition to the protocol structure, the special interaction sequences for the protocol transmitting are also elaborated in the paper. In order to control the robot machine upon the protocol, a particular interpretation scheme is also designed for the bottom control system of the software.

Based on the protocols, the software has been designed and tested; the result shows that all the specific machining function is working smoothly without any interference with one another. The flexibility and the scalability of protocol also enable the modification and reconfiguration of GUIs to be quick and easy in the working field of ITER.

P3-149 **Suppressing chatter methods for robot machines in ITER assembly and maintenance**

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In ITER assembly and maintenance lots of machining tasks need to be done by in-situ machining tools. Those tasks including: welding defectors cutting, biscuits milling, flexible house mining, boring and threading. And all these tasks require carrying out from inside of vacuum vessel of ITER. Normal machine tool like CNC machine is too heavy to work in side of VV. Robot machine is one potential tool for the task. However, as the stiffness is low for serial industrial robot, even a special designed serial robot can hand heavy load over than tons, the great chatter (machine vibration) will still happen during machining process. The chatter vibration affects not only the accuracy and surface quality, but also causes demerge on tools and robot itself.

This paper first investigates the chatter vibration phenomena of flexible manipulators and mechanisms, and then introduces two categories of vibration control, passive vibration control and active vibration control. The passive vibration control focus on structure design, such as the links of mechanisms and manipulators .The active vibration control is the most effective method of suppression vibration of robot. The active vibration control algorithms can be categorized into three basic families according to the controller architecture: the feed-forward, feedback, and hybrid architectures.

In the passive vibration control phase the paper presents a design of parallel kinemics mechanism instead of serial link manipulator to increase the stiffness of the robot machine, and in active vibration control phase the paper studies a hybrid controller that combines feedforward and nonlinear feedback controllers in chatter suppression. The dynamic model of proposal parallel mechanic machine and the analysis of vibration mode of system are given. Finally control results are given based on the prototype of parallel robot machine which was built for carrying out machining and welding for ITER assembly and maintenance.

P3-150 **Divertor Maintenance Scheme for the DEMO Divertor**

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DEMO, as well as ITER, will be Tokamak-type fusion device. Based on the previous PPCS work, the main divertor design principle could be similar to that of ITER; toroidal shape, divertor cassettes and horizontal radial maintenance ports for the divertor. In ITER, divertor plasma facing components are combined as cassettes to make replacement and handling easier. The same principle could be applied in DEMO.

The main differences are:

- DEMO shall demonstrate to viability of fusion for power plant use, so plant availability (thus fast maintenance = short down-time) is an important measure of effectiveness.

- The lifetime neutron fluence on the DEMO in-vessel components is considerably higher than in ITER; therefore the maintenance equipment used in DEMO will have to be tolerant to much higher radiation dose rates than in ITER.

These two differences make DEMO maintenance more challenging than in ITER. The first point requires an optimisation of RH methods and logistics.

The helium-production in steel is due to the high neutron fluence in DEMO even higher than in ITER. As a consequence re-welding of the divertor cooling pipes might not be feasible at the cassette and the divertor cooling pipes may have to be partly replaced together with the divertor cassette.

Due to the high level of activation, the divertor cassette temperature will rise when the tokamak cooling system is disconnected, therefore a maintenance cooling system may have to be arranged during the transportation in the transport cask.

It is recognised that the factors described above will set very different requirements for the maintenance procedures and the maintenance devices. From the maintenance point-of-view, it may be better to increase the number of divertor maintenance ports, ideally avoiding in-vessel operations completely.

This paper will discuss the optimisation of the number of the maintenance ports, the number of the cassettes, number of the maintenance tasks and simplicity of the maintenance operations. The critical issues of the DEMO maintenance are discussed. Proposals affecting the DEMO tokamak design and the maintenance logistics are made.

Topic J Plasma

P3-151 **Study on operation control of helical DEMO reactor FFHR-d1**

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Conceptual design study of the LHD (Large Helical Device)-type helical DEMO reactor FFHR-d1 is now being studied intensively under the Fusion Engineering Research Project in National Institute for Fusion Science. The primary design parameters were fixed and the core plasma design at steady state is in progress based on the direct extrapolation from LHD experiment. Since current ramp-up and steady-state current drive are not needed for the helical system, flexible options for a steady-state operation point and the path toward the point can be adopted. Therefore the design of the heating system (e.g., heating method, maximum power, duration, etc.) and the fuelling system (e.g., pellet size, injection velocity, injection frequency, etc.) can be optimized by the plasma start-up scenario. It should be noted that helical reactors could be started with ECH only and free from the problems related to the NBI port design (e.g., decrease in the blanket coverage, increase in the neutron streaming). The time for the plasma start-up, corresponds to the rate of the change in the thermal output, is also an important factor in the design of the plant equipment, especially plasma facing components. In past study, the plasma start-up scenario of helical system was studied using a 0-D model and a PID control method for the heating power and the fuelling amount was proposed. To optimize the plasma start-up scenario of FFHR-d1, we developed a quasi-1D particle balance model. It utilizes gyro-Bohm-type parameter dependence between local electron density and temperature observed in the LHD experiment and enables a fast calculation of the time evolution of the plasma profile. Using this model, parametric analysis of the start-up scenario with a consideration of the profile effect (i.e., change in the power deposition profile, pellet penetration length, confinement property, etc.). The result will be presented at the conference.

P3-152 **Progress of the ITER EC H&CD Upper Launcher**

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The design of the ITER ECRH system provides 20MW millimeter-wave power for central plasma heating and MHD stabilization. The system consists of an array of 24 gyrotrons with power supplies coupled to a set of transmission lines guiding the beam to the four upper and the equatorial launcher. The front steering upper launcher design described herein has passed successfully the preliminary design review, the following actual step is the development of the final design of the upper launcher. It consists of a millimeterwave system with neutron shielding integrated into an upper port plug with the plasma facing blanket shield module and a set of ex-vessel waveguides connecting the launcher to the transmission lines.

Connected to the transmission lines are the ultra low loss CVD torus diamond windows followed by a shutter valve, a mitre bend section and the feedthroughs integrated in the plug closure plate. These components are connected by corrugated waveguides and form together the primary confinement system (PCS). On the in-vessel side a quasi optical beam propagation via four mirror sets are foreseen with two front steering mirrors. The millimeterwave system is integrated into a specifically optimized upper port plug providing structural stability to withstand critical plasma disruptions

and the high heat load from the plasma side with a dedicated blanket shield module. A recent update in the ITER interface definition consists of a recession of the upper port plug first wall panels, which is integrated into the design. Apart from the millimeterwave system the upper port plug houses also a set of shield blocks which provide neutron shielding. An overview of the actual ITER ECRH upper launcher is given together with some highlights of the design.

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P3-153 **Application of inter-linked superconducting coils for central solenoid and advanced divertor configuration of DEMO**

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Recently, use of an �einter-linked�f (IL) CS in a tokamak fusion DEMO reactor were proposed, in order to achieve a sufficient amount of the CS magnetic flux swing for the current ramp-up with keeping the reactor size reasonable. A basic idea of the IL-CS concept is to wind a CS such that it is linked in a set of toroidal field (TF) coils in order to achieve a large amount of the magnetic flux swing ƒ³cs by increasing the CS cross section, since ƒ³cs is proportional to Rcs^2 where Rcs is the CS radius. We perform engineering analysis of the IL-CS of which outer radius and height were 3.0m and 10m, respectively. The cable-in-conduit conductor like the ITER-CS is selected. We analyze applicability of Nb3Sn and Nb3Al for a superconducting wire of the IL-CS. On basis of this analysis, we chose Nb3Al as superconducting material of the IL-CS. In this presentation, the detailed descriptions of the engineering design, e.g. a conductor structure and a method of winding the superconducting coil linked in TFCs will be presented. Handling of a large exhausted power from the core plasma is the most important issue for the fusion reactor. Recently, advanced divertor concepts of �gsuper-X divertor (SXD)�h and �gsnowflake divertor (SFD)�h, were proposed. The plasma equilibrium calculations for SlimCS showed that large coil currents (~200MAT for SFD and ~70MAT for SXD) are required for the conventional divertor coil location outside TFC. These results show that installation of the divertor coils inter-TFC (�einter-linked�f EF) is required for the DEMO advanced divertor design. In this design, there are engineering issues to be resolved, i.e. severe neutron and thermal shielding for superconducting EFC. In this presentation, engineering feasibility of the inter-linked superconducting CS and EF for constructing the SXD equilibrium configuration will be presented.

P3-154 **Improving magnetic plasma control of ITER plasma**

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The challenge of magnetic shape control of unstable elongated plasma in tokamak fusion reactors is to maintain the prescribed plasma shape subject to large-scale disturbances, such as vertical displacement events (VDE) and edge localised mode (ELM) perturbations, and to considerable changes of local dynamics in different operating points. The plasma current, position, and shape controller design proposal for the ITER tokamak [1] is one state-of-the-art approach to magnetic plasma shape control. It comprises a cascade scheme with an inner vertical stabilization (VS) controller [2], which stabilizes vertical plasma velocity on a fast time-scale, and an outer plasma current and shape controller (SC), which controls the plasma current and the plasma-wall distances (gaps) on a slower time-scale.

In this work we attempt to improve both the control efficiency in suppressing perturbations and robustness to changes of local dynamics by modifying the control scheme so that a faster yet robust response to disturbances is achieved.

[1] G. Ambrosino et al., Design of the Plasma Position and Shape Control in the ITER Tokamak Using In-Vessel Coils, IEEE Trans. Plasma Science, 37, 2009

[2] G. Ambrosino et al.: Plasma Vertical Stabilization in the ITER Tokamak via Constrained Static Output Feedback, IEEE Trans. CST, 19, 2010

P3-156 **Tailoring the optical properties of silica irradiated with swift heavy ions**

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Silica will be the main component of many optical elements in nuclear fusion power plants. In general, these components will receive important doses of radiation and in the particular case of laser fusion facilities the final lenses will have to operate facing the target explosions only a few meters away. In this work, we will concentrate on the permanent effects induced in silica by means of high electronic excitation. In a fusion power plant this can be originated in a variety of ways. For example, energetic PKAs displaced by 14 MeV neutrons, X-ray pulses, ion pulses, laser pulses or swift heavy ions as a result of the target explosion in laser fusion. The evolution of the high electronic excitation leads to deleterious effects such as color center formation, density change, refractive index change, electronic sputtering or surface morphology modification. In this work we will concentrate on the effect of heavy ions with energies exceeding 0.1 MeV/amu. These ions typically generate a high excitation region around their straight passage trajectory that evolves into a nanometer sized cylindrical track. It has been reported that the nano-tracks present higher density than the virgin material for low electronic stopping powers, Se < 7 keV/nm and a low-density core surrounded by a dense shell for Se > 12 keV/nm.

We have studied theoretically (molecular dynamics and optical simulations) and experimentally (irradiation and optical characterization) the dependence of the macroscopic optical properties (i.e., the refractive index of the effective medium, nEMA) on the electronic stopping power of the incoming ions at fluencies low enough to avoid track overlapping. Our results show that the refractive index variation is related to the density change induced by the ion irradiation. We will show that our MD simulations are a powerful predictive tool to simulate the effect of the ions. This could be extended to other sources of electronic excitation such as lasers. As a conclusion, we must emphasize the role of electronic excitation in the degradation of optical components in fusion facilities, up to now not sufficiently studied.

P3-157 **Optimization of Plasma Parameter for Commissioning Scenario without Initial Tritium Loading**

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The commissioning scenario without the initial tritium loading has been proposed. However, the detail control method for core and divertor plasma is not fully investigated. This paper discusses the detail plasma control for the commissioning scenario without the initial tritium loading, in order to optimize plasma operation parameters to minimize the commissioning period to the rating operation phase. In this study, core plasma properties of the MHD stability and the current drive have been investigated for CREST and Demo-CREST. Analysis on MHD stability and current drive were carried out by EQLOUS/ERATO and DRIVER88. In case of Demo-CREST, the gradual increase of tritium density ratio (T-ratio) from T-ratio = 0% is confirmed, keeping the high density operation preferable to the divertor heat-handling during gradual increase of the fusion power in the commissioning phase. An operation route keeping high density by the T-ratio control is also proposed for the commissioning period. Roughly speaking, high temperature operation is preferable to minimize the commissioning period, because of better tritium burning rate to larger tritium production in the blanket. On the other hand, high density operation is required for the divertor detachment. The short commissioning period and the divetor plasma operation condition are in the trade-off relationship. The proposed operation route for Demo-CREST has a consistency with the start-up scenario without the initial tritium loading, and the relationship between the initial tritium inventory and the commissioning period is also evaluated for Demo-CREST. A similar investigation was carried out for CREST, which has a smaller major radius and a more advanced plasma performance than Demo-CREST. Comparison between Demo-CREST result and CREST one will also carried out, and sensitivity of the major radius on the commissioning is analyzed. Finally, optimized plasma parameters and control method for the commissioning scenario without the initial tritium loading is discussed.

P3-158 **Toroidal Alfven Eigenmode Amplifier Control at JET Using Commercial FPGA and PXI Platform to Study Plasma Instabilities**

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At JET a unique 8-coil antenna system has been implemented to study the plasma instabilities due to fast-ion interactions with Alfven Eigenmodes in support of ITER. The present system has a single amplifier with a power of 4kW that is controlled by ageing electronic modules. This single amplifier was driving all eight different in-vessel antennas, preventing the control of relative phase between the antennas, and limiting the signal generation to a single frequency.

The new system is comprised of eight individual 1 kW amplifier to drive the antennas, increasing the total system power to 8kW. The existing function generator and control electronics are replaced by National Instruments FPGA-based cards and PXI Express platform. This unit will control all eight amplifiers in real-time and is also synchronized with the JET system-wide 1 MHz reference clock. The different functions of the digital control unit are to generate the amplifier drive signals with the desired frequency and phase, and to control the gain of all eight amplifiers.

This paper will cover the system architecture, the initial test results and the advantages of this new system compared to the older system.

P3-159 **High-precision power supply for fusion reactor magnets**

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Fusion reactors require high precision magnetic flux to confine the hot fusion fuel in plasma form. The magnetic flux density is directly proportional to current that goes through a conductor, indicating that the magnets power supplies have to be really precise to give the desired magnetic flux. The goal of this project is to design a high performance power supply used to supply these magnets. There have been proposed some improvements on the design of a conventional power supply to meet a high precision current of few tens ppm (parts per million). The system itself consists of two mainly parts, the power and the control system. The power system is composed of buck-converter and a single-phase full-bridge converter with a fifth order low passfilter, while the control system comprises a control board with a 32-bit DSP and a 24-bitresolution Σ-Δ analog-to-digital converter. Furthermore, the power MOSFETs are driven by a high-resolution PWM generated by the DSP that can perform signals with a precision below 1ppm. The control system consists of a current and a voltage loop, both controlled with PI controllers designed after identifying the plant model. Another important issue to take into account is the system immunity to noise sources, i.e. the 50 Hz network and its harmonics. After designing the system, it is experimented by a DSP-based system with a switching frequency of 50 kHz. Several successful tests are made to check its features, including stability, resolution, linearity and repeatability.

Topic K Inertial Cofinements

P3-161 **Ab Initio Calculation of the Speed of Sound for Mixtures of Solid Molecular Hydrogen-Deuterium.**

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In the efficiency of the ignition in inertial confinement fusion one of the critical points is the solid layer uniformity Deuterium-Tritium (DT) target. During the compression process this layer, perturbations grow as the Rayleigh-Taylor instability [1]. If we want to control or to minimize these instabilities is necessary to know the mechanical properties of this layer and its thermo-mechanical limits. We performed a mixture of hydrogen-deuterium (HD) as a first approach to the analysis of DT ice system, where in addition to the instabilities already discussed, is present beta decay of tritium. Through simulation with ab initio methods were calculated elastic constants, the bulk modulus and sound velocity for hydrogen isotopes in solid molecular [2], we have analyzed the phase transitions in this system [3]. Using this methodology we now calculated the bulk modulus and sound velocity for mixing system of hydrogen isotopes, in this case presents the results for hydrogen-deuterium mixtures 50%-50% and 75%-25%, at room temperature and in the pressure range higher than 6GPa. Recent studies show some inconsistencies in the equations of state used in simulations of the fusion process [4], it is therefore necessary a detailed study of the mechanical properties and the equation of state for solid molecular hydrogen isotope mixtures.

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Recent progress in the development of the advanced tungsten alloys in Japan

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Tungsten (W) has a high melting point, high thermal conductivity, and high sputtering resistance; therefore, W and its alloys are considering as one of promising candidate of the high heat flux components (HHFC) in fusion reactors such as ITER and DEMO. During the operation of a fusion reactor, HHFC are subjected to neutron irradiation of 14 MeV and heat flux as high as 10 MW/m2. The neutron irradiation and high heat flux exposure causes embrittlement in tungsten. To suppress the irradiation embrittlement and recrystallization embrittlement, alloying, refining grain structure, and dispersing second phase are the effectual methods. For the application of property improved and advanced tungsten alloys to HHFC in fusion reactor, manufacturing of the materials at the industrial scale is also important. This paper presents the current research status of the development of the advanced tungsten alloys for HHFC application in fusion reactor in Japan, and gives mechanical and thermal properties of the advanced tungsten alloys.