P1.1 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Performance of the TJ-II ECRH system with the new 80kV-50A high voltage power supply

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The ECRH system of the TJ-II stellarator consists of two triode-53.2 GHz-gyrotrons, which can deliver a maximum power of 300 kW each, during 1s. Both gyrotrons are fed by a common high voltage power supply (HVPS) and driven by an anode modulator. During the last experimental campaigns the performance of the gyrotrons and the stabilization of the microwave power were limited by the HVPS, whose maximum output current was limited to 30 A and the ripple level of the output voltage was around 7%. In order to guarantee the reliability of the ECRH system and to improve the performance of the gyrotrons, a new HVPS has been developed and manufactured by the company JEMA and was put into operation at CIEMAT during 2007. The design is based on solid-state technology and high frequency commutation techniques. The new unit reaches -80 kV and 50 A during a maximum pulse length of 1 s. The solution includes a matching transformer, which isolates the AC input and provides the DC current for the 12 pulses SCR rectifier. The DC bus is connected to 32 IGBT invertors, which operate at 2,7 kHz. The pulse width modulated output of each converter is connected to a high frequency transformer, which provides the main isolation from the low voltage to the high voltage side. The square waveform obtained at the secondary of each transformer is rectified by means of a diode bridge. The connection in series of the diode bridges provides the required -80 kV d.c. at the output. In case of arcing in the gyrotrons the HVPS switches-off in less than 5 s, which limits the energy deposited in the gyrotrons and a crow-bar protection is not needed. The level of the output voltage ripple is now 2.5% peak to peak and it will be reduced with an additional filter, which is at present being developed. The complete design, testing and commissioning of the HVPS are presented, as well as the routine operation of the ECRH system during the TJ-II experimental campaign.

Test of a 105 GHz prototype diplexer combiner based on square corrugated waveguide

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Controllable power combination and distribution of multiple sources into multiple transmission lines may increase efficiency and flexibility of ECRH systems. It can be achieved using a new generation of devices now in development [1]. In this work a narrow band diplexer-combiner is presented: It allows fast independent switching of power from two different gyrotrons in two transmission lines. It is based on a resonator made using suitable lengths of square corrugated waveguides (SCW) as beam splitters/combiners [2]. A 105 GHz prototype was installed and tested in a low power configuration. A group of three SCW 2.5 meters long with four mirrors are used to form two nested resonator loops. Signal inserted in the first SCW is equally split in two equal separate beams at opposite angle. One of them is reflected into the second SCW with a mirror. The first loop is closed by placing a second mirror on the launcher side. An identical loop is made with the third waveguide. Two outputs are available and, depending on frequency, the desired amount of output power can be directed in either. Under ideal conditions, a worst-case 98Beam patterns were measured at the output of the first SCW in order to check similarity with the input beam and alignment. Frequency sweeps, performed with a vector network analyzer, were made in order to guarantee that the resonant circuit was working properly and that the theoretical curve was found. This also allowed fine tuning the resonant system. Finally, absolute power measurements were made with a quasi-optical bolometer: It was positioned at the two outputs and the measured values were normalized to the total power entering the SCW. Results of all tests are discussed in this work, together with a feasibility study for insertion of the device in an existing ECRH transmission line.

References:

[1] W. Kasparek, et al., this conference.

[2] A. Bruschi, et al., Fus. Sci. Tech. 53 (2008) 97.

P1.3 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Design of a new ECRH launcher for FTU tokamak

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The present ECRH launcher installed on a single equatorial port of FTU tokamak was designed to inject four beams independently steered continuously in poloidal direction and in a set of predetermined toroidal angles. The launching mirrors can be moved only shot by shot. New experimental programmes for control of MHD instabilities with ECH/ECCD and heating of overdense plasmas with ECBW require new launcher respectively with fast-steerable mirrors and increased toroidal capabilities. The required scanning speed for tracking the rational surfaces in the FTU plasma is 1 cm in 10 ms in poloidal direction, while the maximum toroidal angle needed for O-X-B heating scheme is around +/-40°. Two ECRH lines, feeding the old launcher, will be switched to the new launcher, located in a different equatorial position, capable of launching two independent beams from small movable mirrors in the plasma proximity. A control on the power deposition width will be achieved by changing the beam radius in the plasma using an optical system composed by two mirrors (zooming range 16–25 mm). Place has been reserved for future arrangements of additional components, e.g. a remote steering waveguide. A dedicated feedback control for the poloidal motion of the launching mirrors is being developed, in order to adapt the tracking of the power deposition location to the dynamic changes of magnetic surfaces in real-time. The maximum toroidal angle impacts strongly on the movable mirror design; dimensions (height around 90 mm, width around 54 mm) are limited by the port width (= 80 mm) and the need to preserve the maximum steering angle. Since the mirrors will not be actively cooled, temperature control will be achieved by covering the backside with a high emissivity coating, to obtain an efficient radiative cooling. A detailed description of the launcher is presented in the paper.

A new approach to passive protection against high energy and high current breakdowns in the ITER NBI accelerator grids

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The 1 MV power supply feeding the accelerator of the ITER neutral beam injector (NBI) consists in large equipments whose linear extension exceeds 100 m (direct current generator, output filter, transmission line, high voltage air insulated deck, insulating transformer, high voltage bushing); their capacitances store an energy exceeding by far the energy stored in the present largest NB injectors. As a consequence, the important issue of the arc breakdown current and energy control to reduce the accelerator grids damage becomes crucial for the ITER NBI.

In the present ITER NBI reference design the method to reduce arc breakdown energy and current is still based on the “classical” concept of the concentrated core snubber; nevertheless, due to the large amount of the stored energy, such snubber results a huge component, in spite of the presence of an active polarizing circuit for counter-biasing magnetic core. Furthermore, in the NBI a relatively large part of the HV capacitance to ground remains downstream the core snubber, so that neither the arc peak current nor the high-frequency oscillations can be effectively limited. Last but not least, the concentrated core snubber is ineffective in damping the voltage oscillations caused by insulation failure in any point of the system: in this case, the large voltage reversal associated to the 1 MV dc could determine cascade insulation failure in the most critical components (e.g. HV bushing, transmission line).

In this paper a comprehensive approach to limit the energy-current associated to the breakdown arc and oscillation in the overall circuit is presented. It is based on the concept of a distributed core snubber (DCS), installed along the whole length of the Transmission Line, and on a damper resistor (DR), which connects the last grid of the accelerator (the so called grounded grid) to ground. The main effect of the two components is to increase the arc current oscillation damping: In particular, the DR controls the oscillation of the breakdown arc current, whilst the DCS provides a beneficial effect in reducing oscillation along the whole circuit in case of insulation failure. The structure of the DCS is quite simple: It consists of a stack of coaxial rings coaxial to the 1 MV transmission line central conductor, the inner made of insulated ferromagnetic material and the outer made of carbo-ceramic resistive material; the DR is even simpler, being made by a stack of carbo-ceramic disks.

The paper contains first the assessment of the DCS and DR parameters using a simplified electrical model of the ITER NBI - SINGAP configuration. Secondly, it gives a validation of the DCS concept, implementing a small scale experimental setup together with electrical simulations. The electrical model fairly agreed with experimental results: The structure acts exactly as a R-L parallel circuit where the parameters can be calculated accordingly to geometry and materials characteristics. DCS linearity is assured at least until 80% of the predicted value of saturation flux. Core biasing has been demonstrated to be not necessary.

P1.5 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Detailed design of the RF source for the 1 MV neutral beam test facility

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In the framework of the EU activities for the development of the neutral beam injector for ITER, the detailed design of the radio frequency (RF) driven negative ion source to be installed in the 1 MV ITER neutral beam test facility (NBTF) has been carried out.

Results coming from ongoing R&D on IPP test beds and the design of the new ELISE facility brought several modifications to the solution based on the previous outline design.

Higher power deposition, due to back-streaming positive ions (BSI+) impinging the back plates of the ion source, caused the redesign of most heated components, to fulfil the increased demands on cooling circuits.

Three main cooling circuits are designed to exhaust the heat loads on the RF coils, the source case and the drivers. BSI+ cause very high and localized heat loads and sputtering issues on the vertical inner surfaces of the drivers plate and the back-plate of the Faraday shields. A reference solution for these components based on plasma facing CuCrZr plates and electro-deposited copper with complex cooling channels has been identified; alternative solutions featuring Mo plates, to reduce sputtering effects, are discussed. Thorough thermo-mechanical calculations and fatigue life verifications have been carried out onto these components.

The electric circuit has been designed in detail: Four independent identical circuits are foreseen, each one featuring a series of two RF coils and two fixed capacitors in parallel with other two fixed capacitors, with optimized parameters. The optimum matching with the plasma will be obtained by tuning the RF generator frequency.

The gas supply and the other auxiliary systems have been designed and integrated.

Integration with other components of the beam source has been revised, with regards to the interfaces with the supporting structure, the plasma grid and the flexible connection.

In the paper the design will be presented in detail, as well as the results of the analyses performed for the thermo-mechanical verification of the components.

P1.6 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Vessel design and interfaces development for the 1 MV ITER neutral beam injector and test facility

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In the framework of the design activities for the ITER Neutral Beam Injector and Test Facility (NBTF), the vacuum vessel has been designed concurrently with all the other components, and in particular with the beam source and the large cryopumps, that strongly conditioned the design.

The definition of the interfaces has been focused on the design for the 1 MV NBTF, anyway keeping to the absolute minimum the differences with respect to the ITER NBI vessel.

The vacuum vessel is composed of two separate parts which shall be welded on site: The beam line vessel and the beam aource cessel.

Three main bolted lids are foreseen for horizontal and vertical access to the internal components. The vessel is composed of double wall and ribs in critical areas to minimize the deformations under the atmospheric pressure load.

New concepts for the beam source support, positioning and tilting systems have been developed and an engineering design has been carried out, able to satisfy precise requirements on stiffness, precision of regulation, vacuum compatibility, electric insulation and remote handling operation. These components and the beam source have been fully integrated inside the beam source vessel by means of support structures and vacuum feedthroughs for mechanical links allowing the transmission of motion and forces.

The interfaces between the beam line vessel and the beam line components (BLCs) have been revised to be compatible with the new vessel design and the BLCs support frames.

Further interfaces with the high voltage bushing, the vacuum pumping and the diagnostic systems have been considered. The number and the position of the diagnostic viewports were identified taking into account both diagnostics and structural requirements.

Static, buckling and seismic analyses based on EN 13445 following the design by analysis approach have been performed considering operative and exceptional load cases.

Requirements, criteria and design details are presented in the paper together with the main results of structural analyses.

P1.7 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Design issues of the high voltage platform and feedthrough

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In the ITER NBI system, a high voltage air-insulated platform (named High Voltage Deck, HVD) will be installed to host the ion source and extractor power supply system and the associated diagnostic system. All power and control cables are routed from the HVD to the ion source and extractor via a feedthrough (HV bushing) which connects the HVD to the 1 MV potential central conductor of the gas insulated transmission line. HVD and HV bushing are very special components, far from the present industrial standards as far as voltage (1 MV dc) and dimensions are concerned. Aim of the paper is to focus the insulation and mechanical issues for the two components, making proposals for their construction. As far as the insulation design of the HVD is concerned, the analysis of the electrical field on the external shell surface has been done to optimize the shape with respect to the electric field on the screen and on the surrounding structure, in order to avoid the occurrence of corona/breakdown events. In particular, the design of the surrounding mesh defining the ground potential has been carried out. As for the HVD structural design, after an assessment of the equipment distribution inside the HVD, a preliminary design of the steel and insulating supporting structure has been carried out. On this basis, the seismic action effects on the overall structure have been evaluated by means of a finite element analysis applying the seismic load excitation specified for the ITER construction site (Cadarache); the fulfillment of the seismic criteria according to Eurocode 8 has been verified. As regards the HV bushing, a preliminary design of the complex inner conductor and a solution for the interfaces to the HVD and to the transmission line is proposed and discussed. Concerning the installation, two solutions have been studied under both electrical and mechanical aspects: The first - the reference one-envisages the Bushing installed aside the HVD whilst the second underneath; advantages and disadvantages of the two solutions are presented and discussed.

P1.8 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Preliminary electrostatic and mechanical design of a SINGAP-MAMuG compatible accelerator

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Each ITER NB injector shall provide 16.5 MW auxiliary power by accelerating a deuterium beam across a voltage of 1 MV. At present two possible alternatives for the accelerator are considered: The reference design, based on MAMuG electrostatic accelerator, where the total voltage is graded using five grids at intermediate steps of 200 kV, and the alternative concept, the SINGAP accelerator, for which the total voltage is hold by one single gap. Due to the still pending decision on the accelerator configuration and the possible need of testing both accelerator concepts in the neutral beam test facility, in the framework of EFDA contracts a preliminary integration design of a SINGAP accelerator compatible with the support structure of MAMuG type accelerator has been carried out. The work has been focused on the preliminary electrostatic and thermo-mechanical integration of the SINGAP grids and on the design of electrostatic shields needed to guarantee the voltage withstand capability of the system. In particular, the integration of the electrostatic shields of all the supporting structures and the design of the coolant and electrical lines flexible connections routing from the bushing to the accelerator have been carried out; furthermore, the mechanical review of the post insulator supporting the grid flange and the optimization of their shields has been performed; finally, a shielding arrangement for the highest voltage lines has been proposed. The study is based on electrostatic and mechanical calculations performed by means of 2D and 3D finite element models, with inputs coming from the electrostatic criteria found in literature and from mechanical constraints derived from MAMuG reference design and following review. Moreover a complete modelling of all the components of the beam source assembly by means of new 3D CAD models has been done to demonstrate the feasibility of the proposed design.

Design of a cooling system for the ITER ion source and neutral beam test facilities

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Two test facilities are planned to be built in Padova for the ITER neutral beam injectors: A low energy 60/100 keV Ions Source Facility (ISF) and the 1 MeV Neutral Beam Test Facility (NBTF). The total power to be removed from the two facilities (up to 70 MW) and the power produced by the auxiliary systems are exhausted by the cooling system (CS). The requirements, operational modes and design of the CS are presented in this paper.

Different operating conditions of ISF and NBTF have been considered to identify the maximum heat loads applied to each component of the test facilities. These are the heat loads considered as specifications for the design of the CS.

The test facility components are actively cooled by ultrapure grade water at the specific inlet temperature and pressure required for each group of components. Electrical issues have been also taken into account for the design of the cooling circuits. The use of alternative dielectric coolants has been considered for the temperature control of the plasma grid at about 150°C.

The flow diagram of the CS for the test facilities is presented in the paper. In the heat exchangers of the primary heat transfer system the thermal powers exhausted from the test facility components are transferred to the water of the heat rejection system considering a closed cooling loop for each group of components.

The coolant flow rates and outlet temperatures are measured, acquired and monitored to calculate the heat fluxes and powers. Flow regulating valves could be adjusted in feedback to control the component temperatures in order to fulfil the functional and thermomechanical requirements for each component.

The main design criteria for the CS are the limitations in the maximum material temperatures, the subcooled flow of coolant, limits in the coolant velocities in the cooling channels, manifolds and pipes.

The CS is completed with the chemical and volume control system, vent detritiation and pipeline drying systems.

P1.10 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Assessment of performance of the acceleration grid power supply of the ITER neutral beam injector

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The ITER neutral beam injector (NBI) is designed to deliver 16.5 MW of additional heating power to the plasma, accelerating deuterium and hydrogen negative ions up to -1 MV with a beam accelerated current as high as 46 A. Two main power supplies are foreseen to feed the system: The acceleration grid power supply (AGPS), which provides the power to the acceleration grids, and the ion source and extractor power supply (ISEPS), devoted to supply the ion source components. In the AGPS reference design a step-down transformer feeds an ac/dc converter, which at the output side is connected to five neutral point clamped (NPC) inverters. Each inverter supplies in square wave modulation mode a step-up isolation transformer, which in turn feeds a diode rectifier. The diode rectifiers are connected in series at the output side in order to increase the dc voltage up to the required value (-1 MV for deuterium, -870 kV for hydrogen). Five R-C filters are present immediately downstream of each diode rectifier, to limit the ripple of the output voltage and to limit the over-voltage at beam-off. A -1 MV dc transmission line connects the diode rectifiers to the NBI grids. The AGPS is rated to deliver a total power up to 52 MW to the acceleration grids; each diode rectifier can provide up to 200 kV.

The output performance of the AGPS is subjected to precise specifications in terms of dynamic requirements, voltage ripple, precision and behaviour in case of breakdown of the grids or beam-off. An optimized choice of the main design parameters, like for instance the inverter modulation frequency and the value of the filter capacitances, must be performed as a trade-off between different aspects, with the aim of fulfilling the specifications.

In order to assess the performance of the AGPS, a detailed model of the power conversion system has been set-up and simulated in different operating conditions. The results, which are presented and discussed in this paper, have given very useful indications on the behaviour of the AGPS with different choices of the parameters and on the impact of such choices on the feasibility of the devices. Moreover, the analyses in case of breakdown of the grids allowed estimating the maximum over-current on the inverter switches and to establish a sequence for a safe and reliable protection of the system. Finally, the analyses in case of beam-off provided an indication on the maximum voltage stress at the AGPS output, which is useful to design the dc transmission line.

P1.11 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Progress on the design of the cryogenic plant for the ITER neutral beam injector test facility in Padova

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In the framework of the construction of the test facility for the ITER neutral beam injector (NBTF) to be installed at Consorzio RFX in Padova, a cryosorption cryopump, designed by FZK Karlsruhe, will be employed to guarantee the appropriate pressure range for the D_2 or H_2 pulsed beam. The cryopanels that constitute the cryopump will be indirectly cooled by means of supercritical helium at temperatures ranging from 4.5–6.5 K and a pressure of 0.4 MPa. The thermal shields will be indirectly cooled by means of high pressure (> 1.5 MPa) gas helium at temperatures from 65–90 K. In order to provide the helium gas and the supercritical helium to the cryopump, a cryogenic plant has been further designed by RFX. The cryogenic plant consists of a main refrigerator to supply the supercritical helium, a shield refrigerator based on a Brayton cycle for the cooling of thermal shields, a proximity cryogenic system to provide the helium distribution and a helium recovery and purification system to deliver pure helium to the refrigerators. The plant has been conceived to operate in several functioning modes according to the needs of the NBTF: A cool-down operation, stand-by and NBI operation modes, a 100 K regeneration mode, a warm up operation to ambient temperature, and a 470 K regeneration mode. The full flexibility of the system has been pursued to assure its reliability and to chase unbroken operation. This paper presents the cryoplant design and the technical solutions adopted to optimize the system performances and its capability to implement all the required operative scenarios in a fully automatic mode.

P1.12 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Analytical and numerical models for estimates of power loads

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The total power loads applied to inertially cooled components are commonly calculated starting from calorimetric measurements as cooling fluid temperature increase and flow rate during and after the power load application. In this paper, some generalizations to the standard models of measure interpretation are introduced.

A method is presented to take into account the effect of thermal conduction and radiation between the component and the surrounding material. It consists in evaluating the conduction and convection thermal resistances and the radiation heat flux by numerical models (finite element and computational fluid dynamics methods). An analytical model permits to use these evaluations to obtain, from the measured convection flux, the corresponding fluxes by conduction and radiation.

Sometimes the durations of pulses and duty cycles are such that the calorimetric measurements after one pulse are actually influenced by the previously applied pulses. A correction factor is introduced to cancel this systematic error.

The cooling down phase can be long compared to the duty cycles between the pulses, and consequently the signals cannot be completely recorded till the thermal equilibrium. In order to evaluate the total heat flux on a component also in this case, a model is presented which considers a proper extrapolation curve to estimate the integral of the energy exhausted by the cooling systems.

As an example, the models introduced in the paper have been used to interpret experimental measurements carried out on the RADI experiment at IPP Garching and some results are presented in the paper.

P1.13 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Electrical and thermal analyses for the radio-frequency circuit of ITER NBI ion source

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In 2007 the design of ITER neutral beam injector (NBI) abandoned the arc source of the 2001 baseline, for an ion source of the radio-frequency (RF) type. Four 1 MHz generators are now foreseen, each delivering via a dedicated transmission line a power of 200 kW to a pair of coils inside the source.

In the framework of ITER NBI studies carried out under the European Fusion Development Agreement, this work covers specific thermal and electrical aspects of the radio-frequency circuit for the ion source. Firstly, circuits are presented for adapting the impedance of the load to the impedance of the transmission line, where fixed vacuum capacitors are used and matching is performed by varying the frequency of the generator. Variable capacitors are ruled out due to access constraints and problematic integration with the source layout, an issue critical to the definition of a suitable matching scheme. Possible solutions for the matching components are discussed, in relation to the anticipated equivalent circuit parameters of the RF driven plasma. The role in the matching of the length of the RF line is analysed, over a range of values compatible with installation at the ITER site.

A second issue is represented by the thermal behaviour of the radio frequency transmission line. This must be integrated into a -1 MV dc, 100 m long and SF6 insulated high voltage transmission line, under operational conditions rather different from those of similar existing systems. Suitable coaxial transmission lines of standard dimensions have been considered and compared, studying their thermal performance also with the help of finite element tools.

P1.14 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

A model for electrical fast transient analyses of the ITER NBI MAMuG accelerator

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The design of ITER neutral beam injector (NBI) is based on a five stage electrostatic accelerator, known as multi aperture multi grid (MAMuG) and characterised by an overall acceleration voltage of 1 MV. The MAMuG accelerator requires a large and technologically challenging power supply system, as the multipolar SF6 insulated transmission line (TL) for connection to the acceleration grid power supply (AGPS). Strict load protection requirements apply to the accelerator, subject in operation to breakdowns that must not cause damage to the grids and deterioration of the voltage holding. In order to reduce the energy transferred to the arc, not only fast switching off of the AGPS is required (present specification 150 us from detection of a breakdown) but additional passive protection components are necessary as well, to control the evolution of the discharge on the hundred of nanoseconds timescale of capacitive energy stored throughout the high voltage circuit.

This work presents a complete circuit model for the high voltage circuit of ITER MAMuG accelerator, where careful consideration has been given to the individual components and associated stray capacitance. The 110 m long TL has been modelled with multiple cells and the length of the individual cell chosen coherently with the fastest transients in the system to account for propagation effects. Capacitive and inductive couplings among as many as nine TL conductors are included, on the basis of separate computations with finite element techniques. A scheme is proposed to take into account also the capacitive and inductive coupling of the TL to ground.

The availability of a circuit model at system level is shown to be crucial to a number of design issues, including transient waveforms experienced by the insulation, optimisation of passive protections and of the connections among power supplies, TL and acceleration grids, prediction of the voltage rise to local ground at a distance from the injector grounding point and study of alternative grounding schemes.

P1.15 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Potential failure mode and effects analysis for the ITER NB injector

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The failure mode and effects analysis (FMEA) is a widely used analytical technique that helps in identifying and reducing the risks of failure in a system, component or process. FMEA application requires a system engineering competence and effort, extended to the whole life of the project.

A specific outcome of the FMEA is the assessment of an indicator, risk priority number (RPN), associated to each potential cause or mechanism of failure identified for the system. The RPN can be used to rank the concerns associated to the risk and the necessary corrective actions to be implemented for the system reliability improvement.

The application of a systematic method like the FMEA was deemed necessary and adequate to support the design process of the ITER neutral beam injector (NBI) adapted to the neutral beam test facility (NBTF). The approach adopted was to develop a FMEA at a general 'system level', focusing the study on the main functions of the system and ensuring that all the interfaces and interactions are covered among the various sub-systems. The FMEA was extended to the whole NBI system taking into account the present design status. The FMEA procedure will be then applied to the detailed design phase at the component level, in particular to identify (or define) the ITER class of risk.

FMEA was developed by a multidisciplinary team during a period of one year. Several important failure modes were discovered, and estimates of subsystems and components reliability are now available. FMEA procedure resulted essential to identify and confirm the diagnostic systems required for protection and control, and the outcome of this analysis will represent the baseline document for the design of the NBI and NBTF integrated protection system.

In the paper, rationale and background of the FMEA for ITER NBI are presented, methods employed are described and most interesting results are reported and discussed. Possible protection priorities and strategies are also discussed on the basis of the obtained results.

P1.16 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

The three RF additional heating systems for FAST

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Fusion Advanced Studies Torus (FAST) is a new tokamak machine proposed by the Italian Fusion Associations as a Satellite Tokamak for the ITER programme. FAST has been envisaged to fully accomplish the first three out of the seven missions established by the European Fusion Program (EFP). Missions 4 and 5 are consistent with FAST scientific rationale, while mission 6 and 7 are out of the scope of the FAST conceptual design. Tritium is excluded "a priori"in FAST because, according to the EFP, it is unnecessary for addressing most of burning plasma peculiar physics. Therefore to achieve in FAST burning plasmas conditions, plasma ions will be accelerated in the half-MeV range through an ion cyclotron resonance heating (ICRH) system (f = 60–90 MHz) able to couple up to 30 MW of RF to the plasma. The launchers are based on conventional current straps fed by external conjugate-T matching networks to achieve good resilience to fast plasma instabilities. For long pulse advanced tokamak (AT) scenarios, a 6 MW lower hybrid current drive (LHCD) system (f = 3.7 GHz) has been designed to actively control the current profile. Passive active multijunction (PAM) launchers are envisaged for this system to face the harsh plasma conditions expected in FAST. A back-up solution with a more suitable 5 GHz frequency has been studied, the final choice being affected by the availability of suitable RF power generators (500 kW, CW). The third system is a 4 MW electron cyclotron resonance heating (ECRH) system (f = 170 GHz) that will provide enough power for MHD control. The ECRH power is also available for current profile control and electron heating.

P1.17 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Preliminary analysis of the external matching unit for the ICRH system for FAST

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Fusion Advanced Studies Torus (FAST) is a tokamak machine, proposed by EURATOM ENEA Association, with the aim of preparing ITER and DEMO scenarios and to explore fusion relevant technologies. In FAST peculiar aspect of the physics of burning plasma scenarios will be investigated by accelerating fast ions through additional heating and current drive systems. The main heating source is the Ion Cyclotron Resonance Heating (ICRH) system that will couple to H-mode plasmas about 30 MW of power in the 60–90 MHz frequency range. The launcher is based on an ITER-like antenna made by 8 current straps arranged in 4 poloidal rows and 2 toroidal columns. Six antennas, located in six different equatorial ports, will couple the envisaged 30 MW with a power density of about 10 MW/m^2. The impedance matching system, composed by coaxial components external to the vacuum vessel, includes: Hybrid couplers (splitter and combiner), stubs and phase shifters as adjustable impedance transformer and conjugate-Ts tuned to low impedance by using shifters. Suitable configuration between the output branches of a conjugate-T and straps belonging to different launchers allow arbitrary phasing between straps on the same launcher and should compensate the effects of mutual coupling and asymmetries between them. Two paralleled stubs used in the adjustable impedance transformer improve the performance of the matching system in presence of slow coupling perturbations. The performances of the whole impedance matching system and of the launcher have been carefully evaluated and the results show a good resilience in presence of fast load variations (vswr 1.5:1 at the RF generator level in the worst condition at the launcher-plasma interface). The proposed impedance matching system will therefore minimise the tripping of the power in presence of ELMs.

P1.18 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Past and future upgrades of the gyrotron high voltage cathode power supplies at the Forschungszentrum Karlsruhe

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The high-voltage/high power DC cathode power supply for the gyrotron test facility at the Forschungszentrum in Karlsruhe consists of a 12-pulse thyristor star-point-controller, a 130 kV capacitor-bank followed by a high power tetrode regulator. This system is closely derived from the neutral beam protection system originally designed for JET. It has been procured back in 1983 and has experienced two subsequent upgrades. Originally it was designed for operation at 80 kV/30 A in continuous wave (CW) operation. In 1999 it was upgraded for the development and testing of the 1 MW CW gyrotrons for the W7-X stellerator, to operate at 65 kV/45 amps for pulses up to 180 seconds duration (1:10 duty cycle) and again in 2007 for the testing of 2 MW Gyrotrons at 65 kV/80 amps for pulses up to 10 seconds duration (1:115 duty cycle). These upgrades have been achieved without changes to the main power-bearing components (transformers, thyristor-controller and high power tetrode), which is a tribute to the extraordinary ruggedness the original Siemens-system was built to. The power supply is now 24 years old and is still operating very reliably almost every day.

The paper describes the system, its operation and some critical aspects of the last upgrade, such as the issue of harmonics in the 20 kV distribution mains-grid, the dynamic response of the thyristor-controller and the avoidance of microwave parasitic oscillations in the high power tetrode (CQK-200-4, developed by ABB/ Switzerland and now produced by Thales Electron Devices/France).

Finally an outlook is given on the extension-plans of the FZK Gyrotron-Test-Facility, namely to procure a Pulse-Step-Modulator (PSM) cathode power supply for 100 kV/100 amps, capable of CW operation, for the development and testing of gyrotrons for DEMO with up to 4 MW CW output power.

Prototyping studies for the blanket shield module of the ITER ECH upper port plug

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Within the framework of a concerted action of European associations it is planned to procure ECH upper launcher turnkey systems for MHD control in the ITER plasma. ECH launchers will be installed into four ports on the upper level of the ITER vacuum vessel. They consist of a mm-wave system to inject up to 20 MW mm-wave power into the ITER plasma and a structural system which accommodates the mm-wave components, cooling devices and elements for nuclear shielding. The mm-wave system comprises waveguides, a quasi-optical section, focusing mirrors and a front steering system with a sophisticated pneumatic drive. The structural system is composed of the Blanket Shield Module (BSM), including the plasma facing First Wall Panel (FWP) and the port plug mainframe. A removable flange connection between the BSM and the main frame provides access to the launcher internals. Appropriate remote handling capability is also taken as a design requirement.

The BSM and also the flange connection will be exposed to substantial nuclear heat loads and has to be positioned into the trapezoidal upper port within narrow tolerances. Also baking capability has to be ensured. To meet these requirements a slim wall design has been established. For the BSM and the front segment of the main frame a rigid double wall structure with meandering rectangular cooling channels was designed. Analyses regarding strength, thermo-hydraulic and thermo-mechanical behavior were performed.

To investigate an industrial manufacturing route considering technical and economical aspects as well, two prototypes of a characteristic section of the BSM were manufactured, using two different fabrication techniques. These are

1. hot isostatic pressing (HIP), which combines the sintering of metal powder inside of welded capsules and diffusion welding of solid parts and
2. brazing of bent and machined parts.

The prototypes are under study at the Launcher Handling Test facility (LHT) at FZK, which offers a water circuit to provide coolant with adjustable parameters, simulating different ITER operating conditions. Extensive test series were performed to validate underlying analysis related to homogenous temperature distribution, tolerable pressure drop within the cooling paths and removal of applied heat loads.

This paper describes the design and manufacturing of the BSM mock-up prototypes and first results of the tests performed at the LHT. A comparison between the two manufacturing options is presented with respect to technical feasibility and economical aspects in order to find the preferred fabrication route for the manufacturing of a full size BSM prototype.

P1.20 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

FEM analyses and prototype tests of the UPP structure for the ECRH in ITER

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The ITER electron cyclotron upper port launching system in its present design is fulfilling the main physical goals of local current drive to stabilize neoclassical tearing modes and the sawtooth instability. The flexible mm-wave system is embedded in a rigid upper port plug structure optimized for the rough regular and especially non-regular operating conditions. During regular operation the structure is heated mainly by fast neutrons which are simulated by the Monte-Carlo method. The structural system has to provide sufficient cooling especially in the front part where high heat loads occur. If the active plasma stabilization fails the worst case scenario for the upper port plugs is the upward vertical displacement event (VDE) followed by a fast current quench. The fast plasma disruption occurs close to the plugs; eddy currents are induced and interact with the strong static magnetic field. The resulting mechanical forces and torque moments stress the structural system to its physical limits, a detailed numerical analysis, prototype cross checks and design optimization is required to obtain a working system for ITER. Numerical structural, thermal and fluid dynamic analyses are presented as well as prototype tests, in addition, the structural design is discussed.

This work, supported by the European Communities under the contract of Association between EURATOM and Forschungszentrum Karlsruhe, was carried out within the framework of the European Fusion Development Agreement (EFDA technology task TW6-TPHE-ECHULA). The views and opinions expressed herein do not necessarily reflect those of the European Commission.

P1.21 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Nuclear safety and waste management aspects of the EP ECRH upper launcher for ITER

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An enhanced performance (EP) ECRH launcher is being developed for ITER by several European institutions in a coordinated effort. The main function of the launcher installed in the ITER upper port is to control magneto-hydrodynamic (MHD) instabilities in the plasma. The launcher must have an opening in the blanket, providing free pass-way for the mm-waves, which are coming from long transmission lines and waveguide system. The opening and void space inside the launcher significantly affect the radiation field in the vicinity of the launcher. The resulting radiation loads must meet both shielding requirements and human safety criteria according to the ITER safety regulations. This needs to be proven by means of qualified nuclear analyses in every step of the launcher development.

A system of suitable codes and automated interfaces is used for the nuclear analyses including radiation transport and coupled activation calculations in the proper 3D ITER geometry. Transport calculations are performed by the Monte Carlo code MCNP5 which is coupled by automated interface to the FISPACT activation code. This coupling allows performing shutdown dose rate calculations by means of the rigorous 2 step (R2S) method. The complex geometry of the launcher structure, especially the front steering mechanism, and the need to update on a short term the neutronics model fully consistent to the original CAD design, were the reasons to use the McCad conversion tool for the automated generation of the MCNP geometry models.

The presented work aims at demonstrating that the radiation shielding and human safety requirements set for the launcher and the neighbouring ITER components can be fulfilled. To this end, the effect has been investigated of different sizes and shapes of the shield blocks inside the launcher on the surrounding vacuum vessel, as well as on the superconducting toroidal and poloidal magnets. Waste management aspects for the launcher materials have been investigated for an extended ITER operation up to 20 calendar years according to the M-DRG1 irradiation scenario. The calculated contact dose rates have been used to classify the activated launcher materials according to the French regulations for the radiological zoning scheme to be applied for ITER.

P1.22 (Poster/Topic B : Plasma Heating and Current Drive): Mon, 16:00–18:00 Foyer

Study of plasma formation within the electrostatic residual ion dump proposed for the neutral beam system of ITER

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The electrostatic residual ion dump (ERID) of the neutral beam system of ITER is the object of the present study. This device is composed of five polarized panels of CuCrZr alloy, which deflect and dump the residual D^- and D^+ ions coming from the neutralizer. The 1 MeV beam entering the ERID carries a total current of 40 A, consisting of 60% D^0 neutral atoms, 20% D^- ions and 20% D^+ ions. The main advantage of the ERID design over the more state-of-the-art magnetic RID (MRID) devices is the distribution of the power load among five panels. However, this advantage might be overshadowed by the problem of beam blocking and plasma formation in the narrow channels of about 0.1 m width and 1.8 m length.

In order to characterize the behaviour of the beam in the ERID, we have developed a molecular dynamics code that follows the particle trajectories in the vacuum electrostatic field inside each channel. The deflected-particle deposition patterns obtained using this code show acceptable power densities below 8 MW/m^2. Employing a heuristic approach, we have found that the space-charge effects due to the ion beam separation can be neglected. The deuterium gas re-emission from the panels has been studied with the help of the *TMAP7* code. The results of the 1D calculations indicate that the outcoming deuterium gas flow approaches a saturation level in less than 400 seconds. The outgassing process accounts for approximately one fifth of the base gas density in the ERID channels due to diffusion from the ion source and neutralizer. The two combined gas sources result in a gas target for plasma formation under the impact of the energetic beam particles. The main processes of plasma formation have been analyzed, including the effect of the secondary electrons and the stripped electrons produced in the ERID. The total ionization density rate is found to be two orders of magnitude smaller than the critical value for plasma creation.

The result of this study is that the formation of a plasma is not likely to occur, thus confirming the deflecting properties of the ERID. Nevertheless, some open issues remain, as the role of the stripped electrons from the neutralizer directed toward the ERID channel entrances and the possible influence of the neutralizer plasma.

P1.23 (Poster/Topic C : Plasma Engineering and Control): Mon, 16:00–18:00 Foyer

A real-time current driving control system for the TJ-II coils

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Since the start of plasma operation in the TJ-II stellarator, the required values of the DC currents that are fed to the different coil sets have been driven with high precision along the complete discharge flat-tops. As a result each current configuration produced a highly stable magnetic-field configuration. Recently, the configurational flexibility of the TJ-II has been increased by the commissioning of a new mode of operation that allows magnetic configurations to be varied dynamically during the discharge flat-top. In order to achieve this, new hardware and software features have been added to the TJ-II control system. These new features may also provide new strategies for feedback control in accordance with parameters measured in one or more diagnostics. In this new set-up, coil current profiles are generated and controlled with millisecond timescales by a system based on VMEbus and OS9 real time operating system. A Client-Server architecture using UDP and TCP network protocols has been designed to exchange XML-based data with calling clients. Furthermore, with this software, a fully-functional Java application for supervision and current profiles settings has been developed. This paper provides a detailed description of the complete TJ-II real-time current-profile control system and of the results obtained during its operation.

P1.24 (Poster/Topic C : Plasma Engineering and Control): Mon, 16:00–18:00 Foyer

Advanced MHD mode active control at RFX

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MHD instabilities are actively controlled in the RFX-mod reversed field pinch (RFP) experiment by acting on the magnetic field configuration at the boundary. The active control provides stabilization of resistive wall modes and a significant reduction of the amplitude of error fields and internally resonant tearing modes. By exploiting the real-time MHD mode control system, experimental campaigns with plasma current up to 1.5 MA are routinely executed. Excellent results have been achieved especially on MHD mode suppression at the boundary, toroidal loop voltage reduction, particle and energy confinement time increase and plasma discharge duration. The acquired expertise on the control of MHD modes is relevant for high beta discharges in tokamaks in both theoretical and technological areas. In fact the RFX-mod development exploits advanced control techniques coping with spatial aliasing suppression and decoupling between actuators and sensors. The MHD active control system consists of 192 saddle coils covering the whole torus surface, independently fed. The radial and toroidal components of the magnetic field and the saddle coils currents are acquired and processed in real-time ($3\times192 = 576$ channels). To achieve the results mentioned above, both theoretical tools and technological enhancements have been necessary. A full electromagnetic model of the coil and sensor system in the presence of the passive structures has been developed. It provides the response of the sensor magnetic flux to the applied coil currents and the dynamics of the saddle coil currents driven by the applied voltages. The technological enhancements include field correction coils, fast switching amplifiers, real-time hardware and software. Two new concepts have been introduced at RFX-mod during 2007, both posing new requirements for the system architecture and implying theoretical developments. The first concept is referred to as clean mode control (CMC) and consists in correcting the 2D Fourier components of the raw radial magnetic field measurements. Since the magnetic sensors and the active coils have the same periodicity, the aliasing of the sideband harmonics produced by the active coils entails a systematic error on the Fourier analysis of the measurements. A real time correction algorithm to subtract the sideband effect has been implemented to obtain "cleaned"feedback signals. This scheme allowed a partial phase and wall unlocking of $m = 1$ resistive kink tearing modes ushering in 1.5 MA operation. The second concept is the design of the magnetic field configuration controller according to a multiple input multiple output (MIMO) approach. The single input dingle output (SISO) approach followed so far, though providing a good quality of the static spatial spectrum of the radial field, did not show a satisfactory dynamic behavior. Better results in terms of dynamic "monochromaticity"of the spectrum are expected taking into account the coupling between each sensor and a number of near active coils. The MIMO controller has been developed on the basis of the system decoupling technique. Since the flux transfer function matrix must satisfy the magnetic field solenoidality condition, it is not directly invertible. However, the Moore-Penrose pseudo-inversion is applicable and the design of a pseudo-decoupler was carried out using the singular value decomposition. In order to compute the new algorithms complying with the real-time constraints, the hardware architecture of the active control system has been upgraded introducing faster CPUs and optimizing the data throughput in the real-time network.

P1.25 (Poster/Topic C : Plasma Engineering and Control): Mon, 16:00–18:00 Foyer

Integrated identification of RFX-mod active control system from experimental data and finite element model

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The RFX-mod active system for the control of MHD instabilities has been operating for more than two years contributing to improvements of the discharge quality and length [1]. In the past years a multi-input multi-output (MIMO) dynamic model including active coils and saddle probes has been developed to analyze and simulate the system electromagnetic behaviour, which is strongly dependent on the frequency due to the presence of passive conductors. This model has been derived entirely from experimental data collected ad hoc, by estimating empirically the transfer function matrix which connects the amplitudes of the saddle coil currents to the radial flux measures [2]. Satisfactory results were achieved in reproducing most of experimental responses and the model was used to select and test the gains of operating PID regulators. In view of higher current operation a further increase in control system performance would be desirable and new schemes have been envisaged to take into account the action of active coils on saddle probe measures. To this purpose a dynamic pseudodecoupler based on the dynamic model has been designed and tested in numerical simulations [3]. New measures have been recently acquired in order to further refine the model and to reassess data of critical magnetic couplings, which could affect the reduced accuracy observed in reproducing some flux distributions such as those corresponding to m = 0, low n current patterns. Other useful data could be provided by a 3D finite element electromagnetic model of the load assembly. In order to study the resistive wall modes (RWM) of the plasma in RFX-mod, an adaptation of CARMA [4] model to RFX-mod has been undertaken and a preliminary step for its validation is the comparison of the dynamic response of the radial field sensors. The paper will report on the current state of development of the two models and on their agreement in reproducing the experimental data.

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P1.26 (Poster/Topic C : Plasma Engineering and Control): Mon, 16:00–18:00 Foyer

Nonlinear instabilities induced by the F coil power amplifier at FTU: Modeling and control

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In this paper we focus on the instabilities caused by the nonlinear behavior of the F coil current amplifier at FTU. This behavior induces closed-loop instability of the horizontal position stabilizing loop whenever the requested current is below the circulating current level.

In the paper we first illustrate a modeling phase where nonlinear dynamics are derived and identified to reproduce the open-loop responses measured by the F coil current amplifier. The derived model is shown to successfully reproduce the experimental behavior by direct comparison with experimental data. Based on this dynamic model, we then reproduce the closed-loop scenario of the experiment and show that the proposed nonlinear model successfully reproduces the nonlinear instabilities experienced in the experimental sessions.

Given the simulation setup, we next propose a nonlinear control solution to this instability problem, which is based on the so-called anti-windup compensation paradigm, usually employed to compensate the undesirable effects of the saturation nonlinearity in control systems. In our case, the nonlinearity is not static and is constituted by the nonlinear dynamics of the F coil power amplifier. Nevertheless, the basic idea behind anti-windup compensation still applies and the arising solution is shown to recover stability in closed-loop simulations. Experimental tests are scheduled for the next experimental campaign and will be included in the paper if available.

P1.27 (Poster/Topic C : Plasma Engineering and Control): Mon, 16:00–18:00 Foyer

A new extremum seeking technique and its application to maximize RF heating on FTU

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In this paper we propose a new dynamic extremum seeking technique to be applied to enhance plasma RF power absorption on FTU.

The extremum seeking technique, associated with the real-time plasma position control system, had already proven to be an effective approach to contribute to the optimization of the coupling between lower hybrid radiofrequency antennas and the plasma scrape-off layer (SOL). The technique helped maximizing the percentage of absorbed power and, consequently, the plasma heating, as well as avoiding excessive reflected power and consequently undesired safety shutdowns.

The technique we propose herein is based on a novel theoretical approach, taking into account the nature of existing disturbances affecting the system. As compared to the previous approach, this new technique achieves increased levels of both robustness and performance, thus enhancing the effectiveness of the extremum seeking scheme. The novel approach proposed here is validated on a simulation test-bed arising from identifying the disturbances acting in previous experiments at FTU. In particular, the test-bed is shown to perfectly reproduce the past FTU experiments and the new algorithm is shown to outperform the previous technique when operating in the same experimental condition. The results obtained are very promising, and show that further applications to experiments would lead to interesting scenarios.

P1.28 (Poster/Topic C : Plasma Engineering and Control): Mon, 16:00–18:00 Foyer

Plasma scenarios, equilibrium configurations and control in the design of FAST

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The Fusion Advanced Studies Torus (FAST) conceptual study has been proposed as possible European ITER satellite facility with the aim of preparing ITER operation scenarios and helping DEMO design and R&D. Insights into ITER regimes of operation in deuterium plasmas can be obtained from investigations of non linear dynamics that are relevant for the understanding of alpha particle behaviours in burning plasmas by using fast ions accelerated by heating and current drive systems.

FAST equilibrium configurations have been designed in order to reproduce those of ITER with scaled plasma current, but still suitable to fulfil plasma conditions for studying burning plasma physics issues in an integrated framework. In this paper we report the plasma scenarios that can be studied on FAST, with emphasis on the aspect of its flexibility in terms of both performance and physics that can be investigated. All plasma equilibria satisfy the following constraints:

1. minimum distance of 3 energy e-folding length (assumed to be 1 cm on the equatorial plane) between plasma and first wall to avoid interaction between plasma and main chamber,
2. current density in the poloidal field coils around 30 MA/m^2.

The discharge duration is always limited by the heating of the toroidal field coils that are inertially cooled by helium gas at 30 K. The location of the poloidal field coils has been optimized in order to: Minimize the magnetic energy; produce enough magnetic flux (up to 35 Wb stored) for the formation and sustainement of each scenario; and to produce quite a good field null at the plasma break-down (BP/BT $< 2\times10^{-4}$ at low field, i.e. $B_T = 4$ T and $E_T \sim 2$ V/m for at least 40 ms).

Plasma position and shape control studies will also be presented. The optimization of the passive shell position slows the vertical stability growth time down to 100 ms.

P1.29 (Poster/Topic C : Plasma Engineering and Control): Mon, 16:00–18:00 Foyer

Control, data acquisition, and communication system for the COMPASS tokamak

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The COMPASS-D tokamak from UKAEA is being reinstalled in the IPP AS CR. An efficient operation of the COMPASS tokamak in Prague requires a brand new control infrastructure, including building and machine control, CODAC (Control, Data Acquisition, and Communication), and interlock systems. At present, these systems are beeing build and this paper aims at giving an overview of their status.

The overall plant control includes the building infrastructure (cooling, heating, air conditioning) as well as the machine infrastructure running in the instantaneous service (experimental area access control, cooling water treatment, energetics systems). These services are run continually and they are managed by standard industrial tools. Solutions for a communication with the experiment controller during the experimental run will be presented.

The experiment control had to be developed and tailored particularly for the tokamak operation. On COMPASS will be used a system based on the advanced telecommunications computing architecture (ATCA) and a real time linux running on a PC [1]. This set-up builds a compact CODAC node. Features particular for COMPASS will be described in the paper.

Modelling of the plasma 2D dynamic by means of the MAXFEA code [2] aimed at the development of feedback algorithms will be presented. The derived algorithms will be programmed into the CODAC, which are capable of a real time processing of the acquired signals in a feedback loop well below 50 μs.

Personnel and machine protection will be separated in independent loops. The personnel protection will be based on a restricted access: The interlock will allow the operation of danger systems (high voltage, hydraulic pre-load, high power laser) only with the personnel excluded from the experimental area. Next, the status of the machine systems will be checked during the preparation before each shot. Finally, the shot can be interrupted or operation of particular systems can be inhibited, if necessary. Solutions for the protection systems will be presented.

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P1.31 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Data distribution architecture based on standard real time protocol

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New data distribution architecture has been designed for online data management under long pulse conditions (as the ones required in W7-X and ITER). The distribution system is able to manage both live data and stored data, not only at the local area network level, but also for remote participation environments. The solution can include any kind of signal (waveforms, contour maps or camera movies among others), and any type of formats. From the point of view of scalability, the distributed structure of the architecture and the tree expansion of its elements, enable the system to grow easily based on actual necessities. Other important requirement is security. In this sense, this solution is based on standards, so transmission and access security mechanisms can be applied, and it is full compatible with multi-organization federation systems as PAPI or Shibboleth.

Data distribution is based on real time protocol (RTP) standard, a common protocol for data streaming over IP networks, with important features such as: Timing for data synchronization, flow control, error recovery or session control mechanisms, and because the nature of this protocol, it is full compatible with HTTP. Current implementation frameworks of this protocol enable easy integration of new data formats based on codecs for transmission. It is important to remark the possibility of using indexes associated to data. It enables client systems to retrieve portions of data delimited by timestamps or indexes (for example experiment relevant events).

The presented architecture is designed for data delivering to three main elements: Remote laboratories, processing data elements, and client tools. In all these cases, the access data mechanism is unique, it is based on security policies, and it is valid for stored and live data.

P1.32 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Design, installation and operation of the JET fast visible camera system

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A new fast visible camera diagnostic was installed in JET in September 2006. This diagnostic is part of a diagnostic package in support of the new JET ITER like wall. The scientific objectives of the diagnostic are the study of pellets, ELMs and plasma wall interaction.

The camera is a PHOTRON APX RS with a 1 024 by 1 024 pixels CMOS sensor and capable of acquiring at a maximum speed of 250 000 frames per second. The CMOS technology does not require charge transfer, as opposed to CCD technology, which makes it more suited for operation in a radiation rich environment. Due to very stringent weight limitations no electromagnetic or radiation shielding was possible for the camera. For the same reason, the supporting structure and opto-mechanical components were designed to minimize weight as much as possible. The mechanical, optical, control and data acquisition design aspects of the diagnostic as well as its installation and commissioning will be described.

The camera was operated during JET campaigns C18 and C19 (January-March 2007). The integrated neutron fluence in the camera position after four months of operation is in the range of 4×10^{10} n/cm^2. No evidence of permanently damaged camera pixels or optics was observed. Several disruptions were also recorded, the camera system withstanding field variations of up to 4 T/s. To avoid electrostatic charge accumulation and errors in the data transmission the control program was made to reset the camera and recalibrate its dark signal before every shot.

The camera system provided useful information to study different fusion plasma relevant issues including plasma wall interaction, ELMs and disruption physics. However, the study of ultra-low intensity phenomena, like the continuous plasma wall interaction and the generation and dynamics of impurities requires a higher signal level. It was therefore agreed to expand the original project to allow the addition of an image intensifier to the fast camera system. The upgraded diagnostic is expected to be installed in April 2008. The upgraded mechanical, optical, control and data acquisition design and the acquired knowledge from the installation and operation of the new system will be presented.

P1.33 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Radiation resistant bolometers with Al_2O_3 and AlN substrates, anodized aluminium support frames and improved electrical contacts

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In-reactor tests of JET type bolometers (Au meanders on mica), considered for use in ITER, highlighted several problems: Rapid Au to Hg transmutation, mica swelling, meander detachment, and unreliable spring loaded electrical contacts. To address these important limitations alternative radiation resistant substrates, the use of Pt instead of Au, and more robust electrical contacts are being examined. Basic prototypes with sputtered Pt meander sensors on alumina and aluminium nitride ceramic substrates were tested during electron and neutron irradiation, with results up to 0.013 dpa showing acceptable behaviour of both substrates and better radiation resistance response of Pt compared with Au. However once again electrical connection to the thin Pt meander proved to be unreliable.

Recent work to be reported here, has been devoted to developing new bolometer configurations based on a thin sandwich structure, which facilitate more reliable electrical connections. Different materials for use as the insulating support frame for the delicate sensor substrate have been examined. Initially machineable glass ceramic (MACOR) was considered, and trial supports fabricated to test different configurations and electrical contact designs, however the high boron content makes this material unsuitable for a nuclear environment. Similar support frames were then made from alumina, this being the perfect choice for the application. But alumina was rejected due to cost of machining and fragility. Finally support frames of anodized aluminium were tested. These gave more than adequate electrical insulation for operating temperatures of at least 450°C. Results for the behaviour of the different prototypes as a function of temperature and ionizing radiation will be given, and compared with those for earlier devices. Temperature response for the Pt meander is independent of substrate type and the deposition method. Ionizing radiation at high temperature and dose has little effect on the sensor resistance values, with variations (less or equal to 0.5%) over periods of time far greater than one ITER shot. Several prototypes are now awaiting final in-reactor (BR2) characterization at SCK-CEN, Mol.

A TIEMF model and some implications for ITER magnetic diagnostics

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In recent years it has been clearly demonstrated that radiation and temperature induced voltages and associated currents (generally termed RIEMF and TIEMF respectively) appear in both MI coaxial cables and ceramic coated single cables. While considerable progress has been made in the understanding and even general modelling of RIEMF, the problem with TIEMF (generation of a voltage along the centre conductor of cables due to temperature gradients, particularly large for Cu cored cables) remains unresolved. TIEMF will make it extremely difficult, if not impossible, to separate the radiation and temperature effects from the required signal for the ITER magnetic diagnostic coils.

It has recently been reported that the TIEMF voltages (even without irradiation) are generated in well localized regions of the central conductor, suggesting that some inhomogeneity is present. Optical and SEM examination indicates that mechanical damage, and/or ceramic incrustation in the conductor may be the cause of the inhomogeneities. For operation in ITER, the problem is expected to be even worse, the varying neutron energy spectrum and dose over the whole cable path, will give rise to a non-uniform transmutation (additional inhomogeneity), and induce even in initially perfect cables, TIEMF effects known as RITES (Radiation induced thermoelectric sensitivity).

As with RIEMF, TIEMF is a problem with no easy solution. It must be fully understood and assimilated into the magnetic diagnostics design. With this aim the paper will present a model for TIEMF behaviour from a multiscale point of view, together with some of the expected effects on the EMF induced at the magnetic integrators. Of particular importance is that some geometric configurations expected to be problematic, result in almost zero TIEMF despite the above mentioned effects, and hence provide a more optimistic view of the problem.

P1.35 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Commercial dielectric coated mirrors for ITER diagnostic applications

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In recent years radiation enhanced degradation of reflectivity for different commercial mirrors as a function of irradiation temperature and time (dose) has been studied. The commercially available high quality mirrors being considered for diagnostic and remote handling systems in ITER consist of a thin evaporated aluminium layer on a solid glass substrate. To protect the aluminium surface from damage the mirrors are overcoated with a controlled layer of transparent protective dielectric material (such as SiO and MgF_2) of adequate thickness to obtain optimum optical constructive interference in a given wavelength range. In some cases multiple layers are applied providing enhanced reflectivity over a narrow range.

Here we report the results obtained for different types of dielectric coated mirrors obtained from two manufacturers. They include standard visible to IR, UV enhanced, and multilayer enhanced narrow range ("cold") mirrors. Again good radiation resistance has been observed for all the mirrors tested, with no marked degradation following gamma irradiation up to doses of 50 and 60 MGy at 170°C in dry nitrogen atmosphere. However the problem encountered, as previously reported for other mirrors, has been that the only reliable specification given by the manufacturers is the reflectivity over a given very reduced wavelength range. The commercial mirrors provided as identical in the same batches, have very different reflectivities. Furthermore specifications given for the overcoating materials are completely unreliable. For example XPS analysis of the surface material, has shown F and Th for nominally SiO overcoating, and in the case of UV enhanced mirrors, where the specified coating was magnesium fluoride, neither Mg nor F were found on analysis. As before one must emphasise the importance of measuring reflectivity and analysing composition for all commercial "of-the-shelf" mirrors before use in ITER.

P1.36 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Overview of intelligent data retrieval methods for waveforms and images in massive fusion databases

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JET database contains more than 42 Tbytes of data and it doubles its size about every two years. ITER database is expected to be orders of magnitude above this quantity. Therefore, data access in such huge databases can no longer be efficiently based on shot number or temporal interval. Taking into account that diagnostics generate reproducible signal patterns (structural shapes) for similar physical behaviours, high level data access systems can be developed. In these systems, the input parameter is a pattern and the outputs are the shot numbers and the temporal locations where similar patterns appear inside the database. Therefore, data searches are pattern-driven. These pattern oriented techniques can be used for first data screening of any type of morphological aspect of the signals. It should be mentioned that compound searches can be developed through the use of AND/OR logical operations. Therefore any particular physical phenomenology (for example ELMy or pre-disruptive behaviours) can define the data searching criteria. The training phase of the techniques is devoted to the development of indexing systems with a high capability of data retrieval. This aims at producing results with a minimum expenditure of time (from ms to s) and resources (only a 20% of additional storage is needed). Applications to waveforms (JET and TJ-II) and images (JET) will be shown, where the searching patterns are either entire signals or specific patterns inside waveforms and images. The algorithms already implemented at JET can screen thousands of signals in about one second. Also, searching methods based on the structural shape associated to several physical phenomena (ECE cut-offs and L-H transitions) will be summarised.

P1.37 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Effects of irradiation conditions and environment on the reflectivity of different steel mirrors for ITER diagnostic systems

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First mirrors are critical components for ITER optical diagnostic systems. Due to their position inset in the first wall, but directly facing the plasma they will be in an extremely hostile environment, exposed at elevated temperatures to intense electromagnetic, gamma and neutron radiation, and fluxes of charged and neutral particles. Under such conditions the reflectivity is expected to rapidly degrade. The only viable candidates are metal mirrors, with molybdenum, tungsten, copper, stainless steel, and rhodium all being considered. To date most attention has been given to possible degradation by particle bombardment (sputtering), and deposition (contamination) problems. However first mirrors in addition to gamma and neutron irradiation in high vacuum during operation, will also be subjected to residual gamma irradiation during shut down periods both in vacuum and in an atmosphere of dry nitrogen or air.

In this work possible degradation of the reflectivity for mirrors made from various steels subjected to ionizing radiation, moderate temperature, and different environments has been examined. Mirrors were prepared from conventional stainless steels (316 and 316L) and reduced activation steels (Eurofer, ODS-Eurofer, F82H), and the reflectivity studied from ultraviolet to near infrared, before and after different treatments. The mirrors have been irradiated up to 9 MGy under different conditions of temperature (<170°C) and atmosphere (vacuum, N_2, and synthetic air), as well as heating without irradiation. For the reduced activation steels important reflectivity degradation (<50%) decrease at 500 nm) are observed for the different conditions, and smaller variations (<20%) for the conventional stainless steel mirrors. Surface morphology and microstructure for the mirrors has also been investigated using scanning electron microscopy.

P1.38 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Thermal and mechanical analysis of the ITER plasma-position reflectometry antennas

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The ITER plasma-position reflectometry diagnostic is being designed to provide the distance between the plasma column and the first wall by measuring the electron density profile in the scrape-of-layer up to the separatrix, at four different poloidal locations [1]. Reflectometry measurements are required for correcting or supplementing the magnetic diagnostic system for plasma position control during long pulse operation because of the substantial error of the magnetic coils working under long pulse scenarios (longer than 1 000 s).

This diagnostic consists of four reflectometers that are poloidally distributed from the low field side up to the high field side there being known as gaps 3, 4, 5 and 6. Gaps 3 and 5 waveguides and antennas will be installed inside port-plugs Eq10 and Up01, respectively. However, gaps 4 and 6 waveguides and antennas are to be installed inside the vacuum vessel, this imposing severe design restrictions. In this work we present the relevant results of thermal and mechanical studies carried out for the development of a viable design for the antenna arrangements.

The antennas, located within the poloidal space between blanket modules, will be the components most exposed to the plasma. High thermal loads, due to plasma radiation and neutron nuclear heating, can cause high temperatures and unfeasible stress on certain parts of the antenna structure. Initial constrains on the antenna design arise from thermal simulations results. Because of the vacuum environment, only radiation and conduction heat transfer have to be considered. Once the temperature distribution and the maximum temperatures are kept at acceptable levels, structural analysis has to be performed to know the thermal stress. ANSYS simulations performed using different materials and support structure geometries will be discussed.

Although thermal stress is the most restrictive issue for the antenna designed, it has to be checked that the components can withstand the electromagnetic loads expected during disruptions and vertical displacement events. The stress due to these electromagnetic loads has been calculated analytically as well as with ANSYS simulations.

This work, supported by the European Communities under the contract of Association between EURATOM/CIEMAT, was carried out within the framework of the European Fusion Development Agreement [under EFDA Contract No 06-1439]. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

References:

[1] G. Vayakis, et al., Nuclear Fusion 46 (2006) S836–S845.

P1.40 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Preliminary design of the control and data acquisition systems for the neutral beam test facility

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The neutral beam test facility (NBTF), to be built in Padova, Italy, will develop the ITER heating neutral beam injector (NBI), and will test and optimize its operation up to nominal performance before installation on ITER. The NBTF includes three functional components: The heating neutral beam injector plant aystem (NBI), which is the device under test, the auxiliary plant aystem (AUX), which includes all equipment to operate the NBI in the test facility (e.g. the local electric grid to feed the NBI power supplies), and the NBTF supervisory system that is the electronics/informatics infrastructure to manage the facility. The NBTF supervisory system can be viewed as a 'small CODAC' handling a restricted number of plant systems. The NBI and AUX plant systems will have their own plant control and data acquisition systems (NBI and AUX PCS). The NBI PCS will operate under the NBTF supervisory system during the test phase at the NBTF in Padova and under ITER CODAC after installation on ITER. The NBI is one of the first ITER plant systems required to be ready. It will undergo a testing phase of at least 3 years before integration with the tokamak and its control system is required to be fully compatible with ITER CODAC. For these reasons, it is desirable that the implementation of NBI PCS, the NBTF Supervisory System and ITER CODAC progress in step and in close collaboration. Regarding control, NBI PCS and AUX PCS will have a classical three-layer architecture (plant, cell and device levels) to be implemented by proven industrial technology to provide reliability. The NBTF supervisory system will be a further level on top of the NBI and AUX plant levels. Regarding data acquisition, the injector in operation on the tokamak (ITER NBI) and the one under test in the NBTF (NBTF NBI) will have different requirements. The former shall be a reliable plant system with a minimum of diagnostics to characterize the beam parameters during plasma operation, whereas the latter shall be a well diagnosed experiment to allow R&D aiming at solving theoretical and technological open issues. Hence the diagnostics installed on ITER or NBTF NBI will differ significantly in number and type with a much larger role of data acquisition in the NBTF NBI. The data acquisition needed for the ITER NBI will be part of NBI PCS, whereas the additional data acquisition needed in the NBTF NBI will be part of AUX PCS. MDSplus along with its extension for continuous data collection is being considered for the NBTF data acquisition and management. As the NBI pulse duration will be the same as the ITER pulse (3 600 s), continuous data acquisition techniques will be used. To deal with high sampling rate signals (up to 5 MSample/s), limiting at the same time the data throughput and storage, event driven data acquisition is being considered.

P1.41 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Technology developments for ITER in-vessel equilibrium magnetic sensors

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In-vessel magnetic equilibrium sensors for ITER are being developed at Consorzio RFX in the framework of the EU activities on magnetic diagnostics. Alternative technological solutions for pick-up coil, based on flexible ceramic-coated conductors instead of mineral insulated cables (MIC) as in JET and in other tokamaks, have been considered with the aim of minimizing the measurement drift due to Radiation-Induced and Thermally-Induced Electromotive Force (RIEMF/TIEMF) during long plasma pulses. A number of pick-up coil prototypes have been built using ceramic-coated conductors, potted with fluid ceramic and then tested with the following objectives:

- verify/improve thermal conductivity of ceramic-potted winding pack,
- assess mechanical properties/adhesion of ceramic-potted winding pack,
- assess electrical insulation performances of ceramic-coated conductors,
- assess TIEMF noise in conductors.

The tests have proved that this technological solution is viable, but also that the reliability of the electrical insulation made of flexible ceramic is critical. The thermo-mechanical behaviour of the potted coil can be a critical issue too. For these reasons, further technological solutions have been developed:

- Flexible conductors with high-temperature-resistant braided-fibreglass insulation (P.O.Zh.),
- pick-up coils constituted by a stack of sintered ceramic layers with printed metallic deposit (low temperature co-fired ceramics).

A first set of R&D tests have been carried out on these further design solutions with promising results. The paper describes the design developments, the rationale for the test and discusses the results and the developments necessary for the production of reliable equilibrium sensors for ITER.

P1.42 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Installation and commissioning of the JET-EP magnetic diagnostic system

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A new in-vessel magnetic diagnostic (MAG) system has been developed under the JET-EP enhancement programme, with the aim of improving the accuracy of the reconstruction of the plasma equilibrium and the performance of the real time feedback control of the plasma shape.

The sensors are grouped in two different sub-systems of two field component pick-up coils, assembled on rails in order to ease the installation by means of remote handling. The two sub-systems (upper coils, outer poloidal limiter coils) are attached to different structures of the first wall and replicated for redundancy in two Octants, for a total of 48 new pick-up coils. Most of the coils are made of a mineral insulated cable (MIC) wound around inconel formers, which give a good reliability in the JET environment. The relatively low frequency response (about 50 kHz) is more than adequate for plasma control and equilibrium reconstruction, which require a maximum frequency of 10 kHz. In addition a sub-set of 14 'tangential coils', located on the outer poloidal limiter, is made of titanium bare wire wound onto an alumina ceramic former, so that they can be used for high frequency applications (e.g. MHD studies).

The MAG system was installed in the JET vacuum vessel during the 2007 shutdown and started the acquisition of signals during the restart phase of the machine in spring 2008.

In the first part of the paper the critical aspects of the manufacture of the system will be discussed as well as the in-vessel installation, which was accomplished by means of remote handling tools. The positive results of the commissioning tests show that the new system produces new signals to be potentially used both for real time and off-line applications.

The second part of the paper will be focused instead on the analysis and interpretation of the data collected during the functional commissioning of the system, aimed at the assessment of the physical validation of the new signals. To this purpose the new signals detected during the restart phase of the machine were validated against the prediction of plasma reconstruction codes. The satisfactory conclusion of the functional commissioning will allow the safe inclusion of the new signals in the set of magnetic measurements currently used for plasma equilibrium reconstruction and feedback control of the plasma position and shape.

P1.43 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

New methods of data mining and soft computing for JET with a view on ITER

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In present day magnetic fusion devices, diagnostics can generate great amounts of data; in the last campaigns, JET has produced a maximum of more than 10 Gbytes of data per shot (JET whole data base exceeds 42 Tbytes). On the other hand, due to limited accessibility, the determination of internal parameters relies on measurements of quantities available only outside the plasma, leading to complex inversion problems. New techniques are therefore required to access efficiently the information, to provide a sound statistical basis for the physical studies and to interpret the signals in an unbiased way.

With regard to information retrieval, a significant breakthrough is the development of techniques to store data according to scientific criteria, instead of time intervals and pulse numbers. In particular "structural pattern recognition" algorithms allow selecting, with a cursor on a simple interface, the signal or a subpart of it the user is interest in and then the algorithms produce the list of shot numbers, time intervals and signals in which the same or a similar structure is present. The versions already implemented at JET can screen about a thousand signals in a few hundred milliseconds.

Bayesian statistics is a key element in the integrated data modelling programme, whose objectives consist of making statistically correct use of the information available from a diagnostic set. This approach provides a sound evaluation of the error bars and has already been successfully applied to equilibrium reconstructions at JET, to determine the confidence intervals, typically of the order of a few centimetres, in the main magnetic topology parameters, like the X-point and the magnetic axis position. At the level of analysis and data mining, several "soft computing" methods, like Fuzzy Logic, Regression Trees and Artificial Neural Networks, are being pursued. For example, a Fuzzy Logic identifier has already been developed which can discriminate between L and H mode confinement regimes with success rates exceeding 95%.

For all these topics, detailed results will be presented, based on data of major European devices, and the prospects for ITER will be discussed.

P1.44 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Single crystal CVD diamonds as neutron detectors at JET

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Due the many outstanding properties of diamond in comparison to Si, such as radiation hardness, high electronic grade artificial diamonds grown by microwave plasma chemical vapour deposition (CVD) have been developed since many years to be used as novel neutron detectors in future fusion machines such as ITER and tested at JET. In the latest years major efforts have been devoted to the production of single crystal diamond (SCD) detectors with high spectroscopic performance. So far relevant achievements have been obtained such as 100% charge collection efficiency, high energy resolution (lower than 1%), good detection efficiency and the capability of detecting and discriminating both fast and thermal neutrons at the same time. These features make these detectors highly innovative and promising candidates for future fusion devices with harsh neutronic environment. This paper reports the main results obtained by three new SCD detectors installed at JET for the 2008 machine restart phase. The yields of both total and 14 MeV neutrons produced during plasma pulses have been measured, using three different measuring devices and techniques. The first detector, installed in the vertical port (Oct-1) of JET, is 200 μm thick and is embedded in paraffin. It has a detection efficiency of about $1.0{\times}10^{-05}$ counts/n cm^2 for the 14 MeV neutrons. The second detector, in Oct-1 Limb 1/2, is 104 μm thick and is covered by a thermally evaporated 3 μm thick polycrystalline ^{6}LiF film in order to detect the total and the 14 MeV neutrons. In addition, it is surrounded by an 8-mm-thick polyethylene shield to enhance its thermal neutron response. The third detector is mounted in the main horizontal port (Oct-6) and it is operated in an innovative way, that is with a single low capacitance super screened cable and the whole electronic chain is outside the JET torus hall. Furthermore it uses fast electronics, suitable to the fast diamond response pulse (< 1 ns). It is 75 μm thick, covered by a 3 μm thick ^{6}LiF film and surrounded by a 2 cm thick layer of polyethylene. All the detectors have been previously tested and qualified at the Frascati Neutron Generator. After the description of their main features, the results of the measurements performed are reported showing a very good matching with other classical detectors, such as fission chambers and Si diodes, presently used at JET.

P1.45 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

A web services based system for the distribution of live information in fusion experiments

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In this paper we describe LiveMonitor, an integrated system realized for the distribution of information in fusion environment. The software tool is based on a client-server approach, where the server side consists of a set of web services that collect data from a variety of data sources. Clients can select autonomously the data needed among the information provided by the web services on demand, choosing the request method between synchronous and asynchronous mode. Data coherence among different clients is guaranteed by a refresh mechanism on the server side. A straightforward schema is used for this purpose: A singleton receiver for each web service waits data from a selected source, when the data is ready the receiver collects it and all the web service instances (generated by clients requests) wake up and deliver a copy of the information to the clients.

LiveMonitor has been successfully used at FTU, replacing and enhancing part of the core of the old message broadcasting system. The tool integrates all the information needed by the control room personnel during the experiments, namely the shot sequence status coming from the FTU control system, videos of the plasma discharge from FTU ports cameras, and fresh data from databases. From the hardware point of view, the new system is made of a Linux node running the web-services, while clients running on other machines can display information on large (46") LCD monitors.

The tool has been tested during FTU experiments and can be further expanded to match the needs of control room personnel and experimental physicists.

P1.46 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

The ITER RNC: An updated neutronic analysis

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The Radial Neutron Camera (RNC) is a diagnostic system that will provide the spatial distribution and the total strength of the ITER neutron source (emissivity profile and fusion power) by means of collimated neutron measurements. Line integrated neutron spectral measurements will also provide information on the ion temperature profile. After preliminary work on the RNC carried by ENEA-Frascati in collaboration with ITER International Team, the RNC design has been further improved through the assessment of the neutronic performances using a software tool performing asymmetric Abel inversion of simulated measured neutron signals (MSST) and the Monte Carlo code MCNP. The present design of the RNC consists of two collimating structures for a full coverage of the plasma: 36 collimated lines of sight (LOS) distributed in 3 different planes view the plasma core (ex-plug system) and 9 collimated LOS view the plasma edge (in-plug system).

Neutronic analyses have been performed with MCNP for neutrons and gamma transport calculations through the development of a 3D model of the RNC, in order to optimise the design of the RNC system. The signal due to both DD and DT neutrons, signal to noise ratio (scattered/virgin) and spectra at the detector positions have been calculated. In the present paper the following issues have been analyzed in order to proceed with the RNC design: the possibility of reducing the length of the collimators (resulting in a significant reduction of the overall RNC dimension and weight); the integration of a gamma detection system in the RNC by insertion of neutron absorbing material (LiH) in front of the collimators; the methods for the reduction of the dose due to the neutron streaming through cut-outs in the blanket shielding module. Moreover, the hard variance reduction technique applied to the MCNP source, consisting in sampling neutrons only from plasma regions contributing to the detector signal, has been reviewed and its validity and reliability in the specific problem are demonstrated.

P1.47 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Criteria and algorithms for constructing reliable data bases for statistical analysis of disruptions at ASDEX Upgrade

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The present understanding of disruption physics does not provide mathematical models describing the onset of this instability. Nevertheless, it can rely on a statistical analysis of the acquired data allows to estimate the probability of a disruption to occur. A disruption warning indicator, based on a statistical analysis of the acquired data, is strongly recommended for real time tokamak control system, to take appropriate actions to avoid or mitigate disruption consequences [1]. Disruption avoidance will be of critical importance for future fusion devices, such as the ITER, for which the forces and heat loads caused by disruptions are expected to be orders of magnitude larger than those in present day devices. In the last years, interest has grown on disruption prediction methods. Literature reports several papers where data based procedures are applied to build black box models to estimate the risk of disruptions [1-3]. Generally speaking, data based approaches (e.g. statistical methods, artificial neural networks) provide good solutions to problems where first-principle models are not available. A common issue in data based approaches is the creation of the data base starting from experimental raw data. This is a non-trivial task and data refining procedures have to be performed to obtain suitable datasets. In this paper, several criteria and algorithms for data refining are proposed to build a reliable input set to feed a neural disruption predictor for ASDEX Upgrade (AUG). The final dataset contains disruptive and safe pulses in the shot range 16 000–19 000, selected from AUG experimental campaigns between June 2002 and July 2004. Firstly, 42 plasma parameters for 390 shots (106 safe and 284 disruptive) were selected. A first analysis has been made on the corresponding diagnostic signals in order to identify offset and drift in signal level, missing or faulty values (outliers) and noise. Two main problems have been identified. The electron densities measured by DCN and CO_2 optical interferometers have been analyzed. The CO_2 interferometer integrates the electron density along vertical cords through the plasma core. These measurements are particularly noisy. The DCN interferometer integrates electron density along horizontal cords through the plasma core; these measurements were affected by fringe jumps. An algorithm is proposed to choose, shot by shot, the more reliable electron density signal to be provided as input for the predictor. Furthermore, in this shot range, the locked mode signal was affected by offset and/or drift caused by not compensated magnetic fields from coils. Different algorithms for drift and offset removal have been studied. In the paper, criteria and algorithms adopted to build the data base will be explained. Moreover, the resulting data base consisting of 53 diagnostic signals for 229 pulses (149 disruptive and 80 safe) will be presented. This has benn used in [4] to train, to validate and to test off-line a neural predictor able to promptly detect 92.92% of AUG incoming disruptions.

References:

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P1.48 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Configuration and supervision of advanced distributed data acquisition and processing systems for long pulse experiments using JINI technology

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The development of tools for managing capabilities and functionalities in distributed data acquisition systems is essential in long pulse fusion experiments. The intelligent test and measurement system (ITMS) is a technology that allows the implementation of scalable data acquisition and processing system, based on PXI or CompactPCI hardware, developed by UPM and CIEMAT. In order to use this platform in an extensive way for implementing distributed data acquisition and processing systems, several applications based in JINI technology have been developed. JINI provides a framework for developing distributed applications oriented to services. The applications are based on the paradigm of a JINI federation that is implemented with mechanisms of publication, discovering, subscription and link to remote services. The model implemented in the ITMS platform includes services in the system CPU (SCPU) and peripheral CPUs (PCPUs). So it is possible to obtain the following capabilities:

1. To setup the data acquisition and the data processing to apply to the signals;
2. to provide information about the evolution of data acquisition,
3. to provide information about the data processing; and
4. to detect and to distribute the events detected by ITMS software applications.

With this approach the software applications running in ITMS platform can be understood in terms of implementation details like a set of dynamic, accessible and transparent services. The services search is performed using the publication and subscription mechanisms of JINI specification. The configuration and supervision applications has been developed using remote (LAN or WAN) accessible objects. The consequence of this approach is a hardware and software architecture that provides a transparent model of remote configuration and supervision. So the implementation of distributed data acquisition system with scalable and dynamic local processing capability applied in fusion environment can be simplified.

P1.49 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Automated recognition system for ELM classification in JET

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Edge localized modes (ELMs) are instabilities occurring in the edge of H-mode plasmas. Considerable efforts are being devoted to understanding the physics behind this non-linear phenomenon. A first characterization of ELMs is usually their identification as Type I or Type III. An automated pattern recognition system has been developed in JET for off-line ELM recognition and classification. Contrary to other ELMs analysis techniques, the present method analyzes each individual ELM instead of starting from a temporal segment containing many ELM bursts. The ELM recognition and isolation is carried out using three signals: D-alpha, line integrated electron density and stored diamagnetic energy. A reduced set of characteristics (such as ELM period, diamagnetic energy drop or D-alpha shape) has been extracted to build supervised and unsupervised learning systems for classification purposes. The former are based on support vector machines (SVM). The latter have been developed with hierarchical and k-means clustering methods. The first dataset analysed contained 100 Type III ELMs and 200 Type I ELMs. The success rate of the classification systems is above 85%. Significant efforts are being devoted to testing the robustness of the methods to discriminate other phenomena, like sawteeth or pellets, which could leave a similar footprint in the signals.

P1.50 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

HELOKA data acquisition and control system

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In the framework of Breeder Blankets development for future fusion reactors, the EU is developing Test Blanket Modules (TBMs) that are going to be tested in ITER. In order to qualify the TBMs before their installation in ITER, the EU has planned to test TBM mock-ups and prototypes in dedicated facilities. HELOKA (Helium Loop Karlsruhe) is one of these facilities, currently under construction at Forschungszentrum Karlsruhe, Germany.

HELOKA Data Acquisition and Control System (DACS) controls and monitors the test facility during normal and abnormal operation. Irrespective of DACS, machine protection and personnel safety are ensured by the Central Interlock and Safety System (CISS), through interlock logic acting upon off-normal events. The emergency shutdown system, part of CISS, will secure the safe shutdown in case of severe faults.

DACS is built in stages, synchronously with the helium loop construction. Stage 1 (to be commissioned in April 2008) consists of the data acquisition and control system for the Water Cooling System and the Power Supply. The data acquisition and control for the Helium Loop (including the Helium Turbo Circulator, the Helium Supply System and the Vacuum System), as well as CISS, will be implemented in Stage 2, which is under design. The data acquisition system for the TBM test section and the control of the surface and volumetric heaters are the topic of Stage 3 (under conceptual design). The paper presents the conceptual design of DACS and describes the implementation of Stage 1.

DACS hardware architecture, which is structured into three levels (Supervisory Control System, Subsystem Controllers and Field-Level Equipment) interconnected by buses (Ethernet and Profibus) will be discussed. CISS concept, to be implemented on an independent failsafe controller, will be also described. The software implementation of the State Transition Diagram for Stage 1, which describes the system time-dependent behaviour, will be presented.

The following principles were followed in the design of DACS: commercially available hardware and software, when meeting the requirements, have been used; international standards, wherever applicable, were followed; equipment installed in the field was designed to be self-protecting, rather than relying on network-bound data or remote control for machine protection.

P1.51 (Poster/Topic D : Diagnostics, DAQ and Remote Participation): Mon, 16:00–18:00 Foyer

Perspectives of metal hall sensors for steady state magnetic field measurements in fusion devices

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Magnetic diagnostic based on radiation hard Hall probes is a promising concept for measurement of almost DC magnetic fields on ITER tokamak and future fusion reactors. Good progress has been achieved in development of such sensors based on specially doped semiconductor materials for ITER ex-vessel steady-state magnetic diagnostic. Despite the optimistic forecast for ITER ex-vessel applications, the semiconductor-based radiation hard Hall sensors will probably not be capable to satisfy requirements posed by ITER in-vessel environment (neutron flux about 3 orders of magnitude higher). Alternative option to semiconductor-based Hall sensors could be those based on metals. The supposed higher radiation hardness of the metal based Hall sensors is paid for by somewhat lower sensitivity compared to the semiconductor ones. As a result, metals are only very rarely used for producing commercial Hall sensors and as a result, the knowledge of properties of such sensors is rather limited if any. We undertook initial pioneering feasibility study of use of bismuth (metal with the highest Hall coefficient) as sensing material for steady state magnetic sensors for high radiation environment of ITER and future fusion devices. We will present initial considerations about optimum ratio between the sensing layer size and input current. We will present results of various methods (spring loaded, bonded, joined by high temperature contacting pastas) of providing reliable high temperature and radiation resistant electrical contacts to the thin metal sensing layer. Optimized design of magnetic test stand allowing measurement of Hall sensors properties during temperature cycling from room to 300°C before and after irradiation will be presented. We expect to present also initial results of impact of neutron irradiation with fluence of the order of 10^{17} n/cm^2 on properties of this type of Hall sensors.

P1.52 (Poster/Topic E : Magnets and Power Supplies): Mon, 16:00–18:00 Foyer

Power supply system for the COMPASS tokamak re-installed in IPP Prague

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COMPASS-D tokamak, originally operated by Culham Laboratory of the Association EURATOM/UKAEA, is being re-installed at the Institute of Plasma Physics (IPP Prague) AS CR instead of the "senior"tokamak Castor. Its operation began at the UKAEA in 1989 (in 1992 with the D-shaped vessel). Afterwords, in 2001, COMPASS-D finished its operation in the UKAEA. The scaling factor of COMPASS-D is referential to one tenth of ITER.

The reinstallation of the COMPASS tokamak in the IPP Prague requires a design of a new power supply system. The design of this system has been done during 2007 in collaboration with the ČKD Nové Energo, which has been chosen as a general supplier of this system. Consequently, the individual parts of the system have been manufactured, tested and delivered to the IPP Prague for installation, which has been finished by May 2008 followed by the commissioning phase.

During the Compass pulse the power requirements can reach 60 MW for 1.5 seconds. Two flywheel-generators (47 MVA, 45 MJ, 85 Hz) are installed to supply this power to the local 6 kV grid. Consequently, four modular AC/DC thyristor rectifiers provide the desired current profiles in toroidal and poloidal coil systems. The start-up circuit, which generates the plasma in a tokamak and drives the current in plasma, is based on a fully solid-state switchers and two temporarily inserted resistors. A modular system of converters (only two types of thyristors are used) contributes to the reduced cost and high flexibility of the system. The plasma position feedback system will be supplied using three linear power amplifiers (3×250 kW, 5 kHz) that were designed in the IPP Prague.

In this paper we present the design of the power supply system, as well as installation and commissioning details.

P1.53 (Poster/Topic E : Magnets and Power Supplies): Mon, 16:00–18:00 Foyer

Conceptual design of the quench protection circuits for the JT-60SA superconducting magnets

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In JT-60SA [1], the new satellite tokamak to be built in Naka, Japan, the magnetic field for plasma generation and confinement is produced by a number of superconducting coils whose rated current ranges from 20–30 kA, with a nominal voltage of 5 kV. This paper deals with the conceptual design of the quench protection circuits (QPC) of JT-60SA which have to provide a fast removal of the energy stored in the superconducting coils in case of quench. The core of the QPC units is constituted by a dc current breaker, which diverts the coil current into a resistor for a fast machine de-energization. Different concepts of dc circuit breaker for QPC were considered, including vacuum and static circuit breakers. A hybrid solution, composed of a mechanical bypass switch paralleled to a static switch based on Integrated Gate Commutated Thyristor (IGCT) technology, has been chosen and worked out; a pyrobreaker is added in series to the hybrid switch as a backup protection. The resulting design allows benefiting from the fast breaking and very low maintenance requirement of the static switches, besides maintaining the advantage of the much lower power losses of the mechanical bypass in normal operation. After a brief presentation of the preliminary studies, the selected solution for JT-60SA is presented in detail. The paper discusses the evaluation of the most suitable semiconductors for this application, the selection of the necessary number of devices in parallel, the calculation of the junction temperature evolution in dependence of different parameters (the current to be interrupted and the allowed opening time of the bypass switch) and the maximum stress in case of interruption failure and pyrobreaker intervention.

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P1.54 (Poster/Topic E : Magnets and Power Supplies): Mon, 16:00–18:00 Foyer

Upgrade of the RFX energy transfer system for a reliable 35kV-50kA dc-current interruption

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In RFX the energy transfer system is composed of four identical units and each unit includes a vacuum breaker (two tubes in series), a resistor and a counterpulse capacitor bank. The system is operated so as to divert the magnetising current, initially flowing through the breakers, into the resistor: The consequent flux variation induces a loop voltage in the gas, thus ionising it and starting the plasma current.

In the past, experiments were also carried out at RFX to develop a vacuum circuit breaker unit for the protection of the ITER superconducting coils; the vacuum tubes, of the same type of those used in the RFX energy transfer system, showed a limit in the specific energy I^2t dissipated in the closed contacts: When the threshold of 2.3 GA^2s is exceeded, the probability of having current re-strikes or interruption failures increases dramatically and in some cases even welding of the contacts can occur. During the next experimental campaigns, the current in the RFX magnetising circuit will be increased to 50 kA; simple calculations show that the aforementioned I^2t limit will be exceeded and therefore an upgrade of the system is necessary.

The paper gives a survey of available solutions and discusses the choice made for RFX. Then, the modifications implemented on the plant are described, consisting in connecting in parallel to each existing breaker (main branch), an auxiliary branch identical to the first one, in order to share the current during the loading phase of the circuit, so drastically reducing the energy dissipated on the closed contacts. However, the branches are not operated simultaneously: The auxiliary branch opens first and, after a few tens of milliseconds, the main branch opens, performing the task to interrupt the current.

The performance of the modified system has been tested on one of the four units at the end of 2007; the results are illustrated in the paper and particular evidence is given to the actions taken in order to assure equal current sharing among the two branches in parallel. The power tests, whose results are also given in the paper, were positive and this allowed programming, for summer 2008, the extension of the modifications to the other units of the system.

P1.55 (Poster/Topic E : Magnets and Power Supplies): Mon, 16:00–18:00 Foyer

Towards complete manufacturing of EFDA dipole conductor

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In the aim of a future upgrade of SULTAN facility, EFDA and subsequently Fusion for Energy (F4E), are manufacturing a dipole magnet for testing ITER fusion cables to be performed at CRPP. This task, that started in 2005, is involving Fusion for Energy, ENEA and Luvata (formerly Outokumpu, Italy) for complete developing and manufacturing of the dipole conductor. After the first qualification tests of dipole conductor samples, which have shown performances lower than expected, it has been agreed to manufacture a new set of short CICC with a different cable layout and lower void fraction, successfully characterized. Full manufacturing process required the fabrication of a complete set of dummy CICCs, one for each section of the dipole, for a total length of about 1 000 meters, performed at Luvata, and subsequently sent to magnet manufacturer BNG for assembly of the 1:1 prototype coil. The final conductor fabrication started in the beginning of 2008, and it will be completed by summer 2008. This abstract will explain the steps required to achieve the final cable layout, and the manufacturing process, together with additional studies performed on ancillary aspects.

Neutronic analysis of the JT-60SA toroidal magnets

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JT-60SA is a combined JA-EU satellite tokamak program under Broader Approach (BA) in support and supplement of the ITER program and researches towards DEMO. Several components are foreseen to be supplied by EU including the superconducting Toroidal Field system. The assessment of the radiation fluxes and loads in toroidal field coils (TFC) is basic in their design and could be critical as a significant DD neutron yield is expected. The heat deposited by neutrons and secondary gamma, indeed, if not efficiently removed by liquid helium forced flow cooling, might produce the quench of the superconductors. Thus the evaluation of the nuclear heating is fundamental for the estimation of the temperature margin, the design of the cryogenic system and the quench analysis. Coil insulations can be also affected by radiation damage even if the expected absorbed doses should be below the allowed limit at the machine end-of-life. Moreover, the components activation induced by neutrons results in a relevant safety concern. In the present study a complete neutronic analysis has been performed for the current design of the TFC system. Nuclear heating of the superconducting winding and of the case, absorbed dose of the insulation and neutron spectra have been calculated using the MCNP5 Monte Carlo Code in a full 3-D geometry. Neutron fluxes have been used as input for an activation analysis performed with FISPACT inventory code. Activity, decay heat and contact dose rates have been calculated at different cooling times, assuming a realistic operational agenda. Neutron streaming through the ports, radiation shielding performance of the vacuum vessel and influence of the case thickness have been evaluated and discussed as well as the impact of the nuclear loads on the design of the system.

P1.57 (Poster/Topic E : Magnets and Power Supplies): Mon, 16:00–18:00 Foyer

Toroidal field ripple reduction studies for ITER and FAST

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The amplitude of the toroidal field ripple (TFR) on the plasma operations is one of the main concerns in design high performance tokamaks. In fact it plays a role on the performance of H-mode plasmas and confinement of high energy particles. Two different approaches to control the TFR amplitude will be presented in this paper. The approach currently adopted to reduce the TFR in ITER is based on the installation of ferromagnetic inserts between the vacuum vessel shells. This approach, although reducing the TFR on the plasma separatrix, introduces a limitation in machine flexibility. The same approach has been analyzed in the design of the Fusion Advanced Studies Torus (FAST): A proposal to support the development of ITER and DEMO scenarios in a flexible and cost effective machine able to investigate the peculiar physics of burning plasma conditions in a dimensionless parameter range close to that of ITER. The shape and size optimization of the ferromagnetic inserts has been carried out both for ITER and FAST devices. Details of the systems layout will be given.

A new approach based on the insertion of active coils between the toroidal field coils (TFC) and the plasma, has been extensively investigated for these two machines. This active system would allow reducing the TFR to values even smaller than with the ferromagnetic inserts, without losses of machine flexibility.

All the analyses have been carried out, for both systems, with 2D and 3D electromagnetic FEM codes and showed the possibility to reduce the maximum TFR on the plasma separatrix well below 0.5% both in ITER and FAST by using ferromagnetic inserts. Better results can be obtained by using the active coils fed with a current lower than 1/10 of the TFC current and in opposite direction.

P1.58 (Poster/Topic E : Magnets and Power Supplies): Mon, 16:00–18:00 Foyer

Stress relaxation testing of pre-compression ring mock-up for the ITER magnet system

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The ITER pre-compression ring system is based on two sets of 3 rings fabricated in fibre-glass epoxy composite, located on top and bottom of the inner straight leg region, that reduce the radial outward movement. These rings will provide a radial force of about 70 MN/coil (35 MN at the top and 35 MN at the bottom) under cryogenic conditions pulling the TF coils into contact and reducing the gap between the poloidal shear keys and their keyways during operation. These rings will also reduce the toroidal tension in the four outer intercoil structures, simplifying the design of the insulated joints needed for assembly.

The paper describes the stress relaxation test performed in ENEA Frascati laboratories on a reduced scale glass fibre-epoxy composite ring. The ring was manufactured with a diameter of 1 m (1/5 of the real scale one) by winding S2 glass fibres then impregnated with epoxy. The volumetric glass content was of 69% and no significant presence of defects was verified by an X-ray survey.

The test has been carried out following as close as possible the ASTM standard E 328-86. The ring has been inserted into a dedicated hydraulic testing machine consisting of 18 radial pulling actuators and then loaded up to a radial displacement correspondent to a ring membrane hoop stress of 800 MPa. The constraint has been maintained constant by inserting a stiff steel disk which the ring has been let relax around and the remaining stress has been daily monitored measuring the force required to lift the ring just free of the constraints. The ring performed 120 testing days showing a stiffness reduction of about 2%, mainly obtained within the first 3 weeks and then stabilized for all the test duration.

P1.59 (Poster/Topic F : Plasma Facing Components): Mon, 16:00–18:00 Foyer

Experimental and theoretical investigation of droplet emission from tungsten melt layer

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Tungsten is foreseen as one of armour materials for plasma facing components (PFCs) in ITER divertor and the dome. During the Edge Localised Modes (ELMs) (about 10^4 ELMs per ITER discharge) and the disruptions the armour will be exposed to hot plasma streams. The heat fluxes are expected to be so high that it can cause severe erosion of PFCs thereby limiting their lifetime. During the intense transient events the melting, melt motion, melt splashing and surface evaporation are seen as the main mechanisms of metallic armour erosion. Present knowledge on the material erosion under the high plasma loads is not enough for evaluation of the PFC lifetime for ITER operating scenarios with the disruptions and the type I ELMs.

The plasma loads of ITER transients are not achieved in the existing tokamaks. Therefore other plasma devices such as powerful plasma guns (in particular the quasistationary plasma accelerators (QSPA) are applied for armour testing. To obtain adequate information on the expected damage to ITER PFCs under the transient energy loads the experiments must be supported by numerical simulations using the codes validated against experimental target erosion.

The present work refers to new experimental testing of tungsten targets by plasma heat fluxes relevant to the transient heat loads in ITER in the range 0.5–2.5 MJ/m^2, performed in TRINITI facility. Primary attention is focused at an investigation of melt layer erosion caused by splashing of liquid tungsten droplets. Onset conditions of melt splashing and the properties of the droplets such us size distribution and droplet velocities are studied. Supporting numerical simulations are carried out. Different mechanisms of droplet formation under transient pulsed heat loads are analyzed and numerically investigated. Numerical results on melt splashing are compared with the experimental data obtained at the QSPA-T.

P1.60 (Poster/Topic F : Plasma Facing Components): Mon, 16:00–18:00 Foyer

Optimisation of He-cooled divertor cooling fingers using a CAD-FEM method

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Within the EU power plant conceptual study (PPCS), a modular He-cooled divertor concept [1] is being investigated at the Forschungszentrum Karlsruhe to reach a heat flux of at least 10 MW/m^2. The helium-cooled modular divertor with multiple-jet cooling (HEMJ) has been chosen as the reference design. The HEMJ divertor finger module consists of a tile made of tungsten, a thimble made of tungsten-alloy and a steel cartridge. The cooling is realized by impingement jets of helium (10 MPa, 600°C) through an array of small jet holes located at the top of the cartridge. Besides the jet cooling ability, thermal stresses in the tungsten parts induced by the high heat loads are regarded an important factor that limits the divertor performance and lifetime. Therefore, optimizing the finger geometry for stress reduction is indispensable. In this contribution a combined CAD-FEM method for the optimization of the geometry of the finger components will be outlined.

For the 3D CAD construction CATIA V5 was used following by the thermo-mechanical verifications with the FEM code ANSYS. For the temperature and stress analyses different geometry parameters were varied within a defined range taking into account the HEMJ boundary conditions, i.e. a heat flux of 10 MW/m^2 on the tile top surface, a helium inlet temperature of 634°C, and a heat-transfer-coefficient received from the preceding separate CFD computations. The evaluation results are used as a basis for an iterative feedback for the CAD construction changes.

A design optimized in such a way was successfully high-heat-flux tested at Efremov, St. Petersburg in Russia. The test results already confirmed the expected enhancement of the divertor performance by thermal stress reduction.

References:

[1] P. Norajitra, R. Giniyatulin, N. Holstein, et al., Status of He-cooled divertor development for DEMO, Fusion Engineering and Design 75–79 (2005) 307–311.

P1.61 (Poster/Topic F : Plasma Facing Components): Mon, 16:00–18:00 Foyer

Current status of He-cooled divertor development for DEMO

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A He-cooled divertor concept for DEMO [1] is being investigated at the Forschungszentrum Karlsruhe within the framework of the EU power plant conceptual study. The design goal is to resist a heat flux of 10 MW/m^2 at least. The major R&D areas are design, analyses, fabrication technology, and experimental design verification. A modular design is preferred for thermal stress reduction. The He-cooled modular divertor with multiple-jet cooling (HEMJ) was chosen as reference concept. It employs small tiles made of tungsten, which are brazed to a thimble made of tungsten alloy W-1%La_2O_3. The W finger units are connected to the main structure of ODS Eurofer steel by means of a copper casting with mechanical interlock. The divertor modules are cooled by helium jets (10 MPa, 600°C) impinging onto the heated surface of the thimble.

In cooperation with the Efremov Institute a combined helium loop and electron beam facility (60 kW, 27 keV) was built in St. Petersburg, Russia, for experimental verification of the design. It enables mock-up testing at a nominal helium inlet temperature of 600°C, an internal pressure of 10 MPa, and a pressure drop in the mock-up of up to 0.5 MPa. Technological studies were performed on manufacturing of the W finger mock-ups. Several high heat flux tests were successfully performed till now. Post-examination and characterisation of the mock-ups subjected to the HHF tests were performed.

Altogether, the test results confirm the divertor performance required. The helium-cooled divertor concept was demonstrated to be feasible. The knowledge gained from these experiments and some aspects on the design improvement are discussed in this contribution.

References:

[1] P. Norajitra et al., He-cooled divertor development for DEMO, Fusion Eng. Design 82 (2007) 2740–2744.

P1.62 (Poster/Topic F : Plasma Facing Components): Mon, 16:00–18:00 Foyer

Development of a He-cooled divertor: Technological studies on W machining

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A He-cooled divertor concept for DEMO has been pursued at Forschungszentrum Karlsruhe within the framework of the EU power plant conceptual study since 2002. The design goal is to achieve a DEMO-relevant heat flux of 10 MW/m^2 at least. The current reference concept He-cooled modular divertor with jet cooling (HEMJ) relies on impingement cooling with high pressure helium (10 MPa, 600°C). This modular design is favourable to reduce thermal stresses. It employs small tiles made of tungsten, which are brazed to a thimble made of a kind of tungsten alloy W-1The major development work areas are design, analyses, material issue, fabrication technology, and experiments for design verification. The choice of tungsten as divertor material is based on its high resistance against sputtering. It is a low-activating material and possesses a high melting point, high thermal conductivity, and relatively low thermal expansion. On the other hand, it possesses high hardness, and a high brittleness, which make the fabrication of tungsten components comparatively difficult. Machining tungsten is therefore a challenging task when aiming at

1. high surface quality in connection with minimising the (micro)-cracks potentially initiated by cutting procedure, and
2. economic point of view, i.e. possible employment of mass production process.

For the purpose of experimental verification of the design, a helium loop was built at Efremov with an electron beam facility installed for high heat flux (HHF) tests. The divertor mockups were manufactured. The tungsten parts (tile and thimble) are generally machined out of a full rod. Conventional cutting processes like turning, milling and electrical discharge machining (EDM) were applied. The latter was found to induce micro-cracks on the tungsten surfaces. In this contribution the fabrication technology especially the machining of tungsten divertor parts will be outlined and the potential of mockup quality improvement discussed.

P1.63 (Poster/Topic F : Plasma Facing Components): Mon, 16:00–18:00 Foyer

Experimental and numerical investigation of a 1:1 mock-up cooling finger of a helium-cooled divertor

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A helium-cooled divertor concept (HEMJ) for fusion reactors of the generation after ITER is being developed at the Forschungszentrum Karlsruhe [1]. The reference design is based on the impingement cooling technology, where small multiple jets of high-pressure helium (10 MPa) are directed onto the hot surface to be cooled.

Thermohydraulic simulations with the commercial fluid dynamics codes ANSYS FLUENT and ANSYS CFX predicted unusually high local heat transfer coefficients (up to 50 kW/m^{2}K [2]). To validate these results, experiments were carried out in the Helium Blanket Test Loop (HEBLO) with a 1:1 mock-up made of steel and brass. The thermal conductivity of brass (119 W/mK) is close to that of tungsten used for the cooling finger reference design (124.6 W/mK at 650°C).

The moderate heat loads simulated by an electrical heater (1 or 2 MW/m^2) and mass flows (1.2–6.1 g/s) were varied. The pressure loss and the temperature distribution in the mock-up were measured. Operation points were then simulated with the commercial software tool ANSYS CFX.

The agreement between experiments and simulations in terms of the temperature distribution is acceptable. The pressure loss is underestimated by the code by about 20%. The results can be extrapolated to the cooling situation of the real geometry of the divertor plates. Under DEMO operation conditions (inlet temperature 634°C, inlet pressure 100 bar), a pressure loss of a cooling finger of about 1.5 bar is predicted.

In this contribution, the experimental and simulated results will be presented and discussed.

References:

[1] P. Norajitra, A. Gervash, R. Giniyatulin, et al., He-cooled divertor development for DEMO: Status of design and HHF tests, Proceedings of the ISFNT 08, Heidelberg, Germany, Oct. 2008, Fusion Engineering and Design, under review.

[2] T. Ihli, R. Kruessmann, I. Ovchinnikov, et al., An advanced He-cooled divertor concept: Design, cooling technology, and thermohydraulic analyses with CFD, Fusion Engineering and Design 75–79 (2005) 371–375.

P1.64 (Poster/Topic F : Plasma Facing Components): Mon, 16:00–18:00 Foyer

Molecular dynamics simulation of surface vaporization in plasma facing components under neutron exposure conditions

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The aim of this work is to understand on a physical basis the main features of plasma facing materials surface vaporization under typical tokamak conditions. In particular the synergetic effect of high vacuum and neutron exposure has been investigated. Plasma facing components behaviour is unique in that these components are kept in high vacuum conditions and at the same time are exposed to high thermal fluxes due to plasma and neutrons exposure. As a consequence the resulting evaporation rate can affect the overall tokamak performance, by plasma dilution and contamination, and can reduce plasma facing components lifetime by erosion. Surface sublimation of light plasma facing materials like beryllium can be affected by lattice defects, ad-atoms migrations and formation at the surface. However an important issue not yet deeply investigated is the potential enhancement of surface vaporization due to neutron exposure. Within a large tokamak like ITER and future fusion reactors a not negligible fraction of neutrons produced by D-T reactions will interact with plasma facing components. Neutron flux exposure will give rise to long term effect, by changing physical and mechanical properties of beryllium. Moreover it could also enhance plasma facing materials vaporization, by increasing beryllium bulk temperature and removing atoms from equilibrium lattice positions. In order to understand the underlying physical mechanisms and to evaluate the main parameters affecting plasma facing component behaviour, a mixed method of analysis based on coupled Monte Carlo and molecular dynamics simulations has been employed. First a number of Monte Carlo runs have been performed in order to evaluate neutron absorption within plasma facing components thickness. Then molecular dynamics simulations have been carried out, giving the appropriate initial speed to neutron knocked-on atom. In this way an estimate of lattice damage produced by neutron exposure has been provided. The modified lattice configurations resulting from MD simulations have been taken as starting conditions for subsequent molecular dynamics simulations, aimed at evaluating beryllium surface vaporization. Results show that neutron exposure can have not negligible effects on plasma facing components thermal properties and evaporation rate.

P1.65 (Poster/Topic G : (In)-Vessel Engineering and Remote Handling): Mon, 16:00–18:00 Foyer

Thermal and hydraulic analysis of the cooling system for the ITER equatorial port plugs

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Engineering activities related to the ITER port plug diagnostic integration are aimed to provide analyses not only for engineering design and thermal-mechanical calculations, but also for assessing QA requirements, as well as for the development of standards and guidelines corresponding to the design engineering and integration. Within this framework, EFDA has launched several activities. Amongst these there is one related to the thermo-hydraulic analysis of the equatorial port plug (EPP) which is currently being performed by CIEMAT. The starting point is a draft design of the equatorial port plug cooling circuits, considered as a "reference layout", which will be modified in accordance with the results from the thermal-hydraulic analysis in order to fulfill the cooling/baking requirements.

In the current study two different analyses have been carried out. The first one is focused on calculating the main hydraulic parameters for the reference circuit, analyzing different piping configurations in order to achieve a balanced circuit in terms of the mass flow of coolant diverted to the different branches, this being a key point for obtaining a homogeneous refrigeration of the port plug. The calculations of this analysis have been performed using ANSYS fluid 116 element, a 3D pipe element that simplifies the model and therefore makes it possible to introduce as many changes as needed to get an optimized circuit.

The second part of the study is focused on the cooling requirements for the different design options proposed by UKAEA and CEA (associations within the above mentioned design engineering and integration tasks) for Diagnostic Shield Modules (DSM). The integration of these DSM requires significant changes to the equatorial port plug cooling system, in particular for the UKAEA design as it entails a port plug cooling based on thermal contact between the DSM and the port plug, whereas in the reference design the EPP cooling is based on the piping cooling circuit running all over the different parts that need to be cooled. For the calculations of the thermal-hydraulic behaviour of the diagnostic modules a more powerful CFX code has been used, in addition to the ANSYS fluid 116 element, due to the complexity of their geometry.

The above analysis means not only an improvement in the equatorial port plug cooling system design, but also a better understanding of the whole system in order to foresee the weak points that will have to be taken into account in the subsequent detailed design.

P1.66 (Poster/Topic G : (In)-Vessel Engineering and Remote Handling): Mon, 16:00–18:00 Foyer

Feasibility study of the cut and weld operations by RH on the cooling pipes of ITER NB components

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The maintenance operations of ITER NB components inside the beam line vessel (BLC) involve the removal of the faulty component, its transport to the hot cell as well as the reverse operations of transport of the repaired/new component and its reinstallation inside the vessel. Prior to the removal of the BLCs the cooling pipes must be detached from the component following a procedure that applies to the cutting of the pipes and subsequent welding when the component is re-installed. The purpose of this study, conducted in the framework of EFDA, is to demonstrate the feasibility of the cut and weld operations on the water (and gas) pipes of the BLCs using fully remote handling techniques. Viable technologies for the cut and weld operations have been identified within the study. In particular the following aspects will be presented in the paper:

- Different strategies can be pursued in the detachment of the components depending on the number of cut/weld operations to be performed on the pipes. The selected strategy will impact on the procedure to be followed likewise on important aspects as the requirements of the flexible joints (bellows) assembled on the pipes. A comparison of the different alternatives will be discussed in the paper.
- The existing cutting techniques have been examined in the light of the remotely performed pipe cutting at the NB cell. Modifications of commercial tools (considering the lathe cutting method) have been proposed in order to adapt them to the BLCs pipes requirements.
- The debris produced during the cutting process must be controlled and collected, therefore a cleaning system has been integrated in the adapted cutting tool referred above.
- The existing welding techniques have been also examined and compared based on different criteria such as complexity, reliability, alignment tolerances, etc. TIG welding is the preferred technique as it stands out for its superior performance. The commercial tools identified need to be adapted to the NB environment.
- The alignment of the pipes is a critical issue concerning the remote welding. A proper alignment system has been proposed taking into account the pre-selected welding technique. This system is also compatible with the cutting tool.
- In order to avoid cut/weld on the neutralizer gas pipes for their connection/disconnection, a solution based on removable connections has been proposed. The system basically consists of a self-aligning device which integrates commercial quick-acting connectors.

P1.67 (Poster/Topic G : (In)-Vessel Engineering and Remote Handling): Mon, 16:00–18:00 Foyer

Design of an overhead crane for the ITER NB cell remote handling maintenance operations

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In the neutral beam cell of ITER all the maintenance operations on the neutral beam components (BLCs) must be performed by remote handling means. The maintenance operations involve the removal/reinstallation of the faulty/repaired component and its transport to the hot cell door. Since a top access strategy has been adopted, all these operations must be performed by an overhead crane of large payload capability (30–50 tonnes). The crane system consists basically of a monorail, a carriage, and a lifting mechanism. The monorail solution is dictated by the NB cell structural layout, with the pillars and bioshield severely limiting the space availability. One tight constraint is the height of the NB cell ceiling that cannot be modified. The crane must access the BLCs in the beam line vessel, the front components connecting the NB vessel with the tokamak, and a storage area at the north end of the NB cell. In the present study, a design of the monorail layout and the crane system has been developed, that allows to perform all the required maintenance operations within the strict spatial constraints of the NB cell. In parallel with the crane design, a logistics and space availability study has been carried out, leading to the detection of clearance or transport problems that could be taken into account in the final crane design. The final monorail layout has four radial branches (one above the symmetry axis of each injector), one toroidal branch to service the front components, and one branch running parallel to the north wall of the NB cell, to access the storage area and help transporting tools required for the beam source maintenance. Rail switching points are required at the intersections between radial and toroidal branches. A translational switching mechanism is proposed. The crane carriage consists of two independent subcarriages, each composed of four wheels. A set of four secondary wheels attached to the main carriage prevents the crane tilting due to the CoG misalignment of some loads. The elevation system needs to handle large loads with great accuracy and stability. A four-rope solution has been chosen for its greater operational flexibility and accuracy at component installation. Two elevation mechanisms are compared: One electromechanical and one hydraulic with the same operational capabilities.

P1.68 (Poster/Topic G : (In)-Vessel Engineering and Remote Handling): Mon, 16:00–18:00 Foyer

IFMIF target assembly: Enhancement of the remote handling strategy for the replacable backplate bayonet concept

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One of the most technically challenging activities of the IFMIF facility is the maintenance and the refurbishment operations of its components, and among these the target system appears to be critical since it is located in the most severe region of neutron irradiation (50 dpa/fpy). The reference IFMIF target design is based on the use of a replaceable backplate. This choice was originally conceived to reduce the waste material, to increase the overall target system lifetime and to improve the duty cycle of the IFMIF plant. Two different backplate concepts have been developed: The backplate based on the cut and reweld concept and the bayonet one. These two concepts differ from the mechanical point of view as well as for the remote handling (RH) strategy to be used for their replacement: The cut and re-weld concept has been designed to be replaced in a hot cell, whilst the bayonet concept was devised to enable the in-situ replacement. This latter concept has been tested many times and its suitability to remote handling was already proved. Nevertheless the present RH procedure developed for the replacement of the backplate foresees the removal of the whole components from the upper part of the test cell (i.e. vertical target assemblies (VTAs) and of the seal plate of the test cell cavern) to allow access to the backplate. This operation has a strong impact on the intervention time for the backplate refurbishment, and since it has to be repeated at least every year (the backplate life time is expected to be of about 11 months, but it is still under assessment) the need to update the RH strategy for the refurbishment of this component becomes a precondition to fulfill the stringent requirement to enhance the duty cycle of IFMIF plant. Several proposals to review the RH strategy for the backplate replacement are under discussion within the IFMIF community all entailing modifications of the IFMIF test cell design. In particular two approaches are presented in this paper: The first relies on the availability of a lateral window on the test cell cavern and the second one foresees the modification of the upper part of the test cell itself. Advantages and disadvantages of these approaches to enhance the RH strategy for the substitution of the backplate bayonet concept are also reported.

P1.69 (Poster/Topic G : (In)-Vessel Engineering and Remote Handling): Mon, 16:00–18:00 Foyer

Neutronic analysis of ITER cryopump system

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The ITER vacuum vessel has upper, equatorial and lower port structures. The bottom ports are dedicated to the divertor components replacement (5 ports) and to the vacuum pumping by means of cryopumps (4 ports). The latest cryopump port design is more complex as it has a pump with a direct view of the vessel (upper cryopump) and a second pump at the end of a branch port (lower cryopump).

3D neutronic analyses have been performed in order to study the radiation conditions in and around the port system. In detail nuclear heating on the cryopump has been calculated updating previous analysis performed in 2003 [1]. Calculations have been performed by means of MCNP 5 Monte Carlo code supplied with FENDL 2.1 library. In this work a new 40-degree model of ITER has been used in which full details of the cryopump system and remote handling ports have been included as well as the updated divertor components.

The paper will present the neutronics results consisting in nuclear heating and neutron fluxes on cryopump components, divertor port outer components such as the dummy rail and water tubes feeding the cassette. Dpa and helium production estimates are provided as well in zones where welding is a critical issue.

Neutron streaming through the ports is analysed and compared to the main streaming sources e.g. the divertor cassette gaps which in the reference design are 10 mm wide, but an increase to 20 mm is being contemplated.

Gamma doses after shutdown have been calculated around the port flange to have an idea of the possible dose to which the eventual operator will be subject and to plan adequately manual operations. Dose around poloidal field coil 5 near to the port has been calculated too as it is an access point for coil maintenance.

The cryopump is located at a distance of 5 meters almost from the mouth of the divertor port and it is 3 meter long. Calculations of such deep penetration problem are very challenging requiring special variance reduction techniques with Monte Carlo codes in order to use in an efficient way the computer resources. These will be described.

References:

[1] L. Petrizzi, ITER CTA Detailed Neutronic Analyses, Final Report on contract EFDA/01-633 ENEA ref NE-VV-R-001 April 2003; also included in Nuclear Analyis Report NAR ITER ref document G 73 DDD 2 W 0.2 (v2.0) March 2006.

P1.70 (Poster/Topic G : (In)-Vessel Engineering and Remote Handling): Mon, 16:00–18:00 Foyer

Mechanical testing of an FW panel attachment system for ITER

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Two options have been considered for the attachment of the first wall (FW) panels onto the shield block of the blanket modules for the ITER 2001 design. Option A is based on direct access bolting and consists of a central poloidal key at the panel rear side which bears poloidal and toroidal forces and radial moments. The panel is attached by means of ten special studs located on a key-way in the bottom shield block which bear radial forces and poloidal or toroidal moments. An objective of the tests was to check the stiffness, strength limit and fatigue behavior of the attachment system (AS) under loads simulating conditions during off-normal plasma operations.

Special device for a test of stud tensile pre-load relaxation during 30 000 temperature cycles between 100–200°C were developed. Two methods were used for a separation of the real pre-load drop due to the thermal behavior of the thread in the panel and a false drop induced by a strain gauges creep.

Four panel mock-ups (1080×250×50 mm) and massive shield block having all the features of the real AS were fabricated from 316L stainless steel. The panel screwed to the shield with the stud pre-load from 45–100 kN was then loaded alternatively by 2 500 cycles of radial moment (±24.5 kNm), poloidal force (±108 kN) or poloidal moment (±53 kNm) at RT. Stud bending stresses, the stud pre-load relaxation, the cyclic deformation leading to the undesirable gap opening at key-way and possible plastification of AS were studied. FE models were used for optimization of test rig as well as for complex interpretation of results. FE simulations were used also for an estimation of combined loads effect during a single off-normal event such as VDE or halo currents.

Results have shown that the thermal cycling leads to the stud preload drop from 100–60 kN whereas the mechanical cycling itself does not cause the additional loss of the pre-load. Applied loads also do not induce the loss of the firm vertical contact in the key way or damage of AS nor under the low pre-load of only 54 kN. The small vertical gap was observed only under poloidal moment with extremely low pre-load of 45 kN.

The combination of poloidal moment and radial force during VDE seems to be the most dangerous case because it could lead to the loss of the contact between the panel and shield block even under relatively high stud pre-load. This FEM result should be verified by means of additional experiments.

P1.71 (Poster/Topic G : (In)-Vessel Engineering and Remote Handling): Mon, 16:00–18:00 Foyer

Electromagnetic analysis of the new ITER blanket modules for vertical passive stabilization

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A new blanket module design proposed recently for ITER is made up of two different parts: The First Wall (FW) panel and the Shielding Block (SB). The FW panel is finely sliced in the poloidal direction and it is made by three different materials: Be, CuCrZr and 316 LN SS. These materials are subject, during operation, to different temperatures and therefore different resistivities. The SB is made of SS with the lowest temperature being the component farthest from the plasma region. Finite element analyses with the ANSYS code have been performed simulating a downward VDE disruption with a plasma current quench of 16 ms (exponential) time constant by extracting data from the corresponding DINA file, see [1]. A routine has been developed for transferring data from the DINA files to the ANSYS mesh which allows modelling the actual plasma current profile and position versus the time. This routine can be conveniently used for all kind of disruptions described by DINA input files. Several configurations have been analyzed, considering both the blanket module electrically insulated (this case is considered as a reference) and all toroidal rows of blanket modules (except for those corresponding to the upper and equatorial ports levels) electrically connected. In particular the solution with a single strap placed in the middle equatorial plane of each blanket and the solution with two symmetric straps respect to this plane have been analyzed. Moreover the influence of the radial position of the straps has been evaluated. The analysis of the results obtained has shown that the general effect due to the electrical straps between modules, if compared with the scenario without straps, is to reduce the torque and to increase the net force applied to each blanket. As far as the radial torque is concerned, the option with two straps is better than the option with one strap. The effect of moving the straps to the plasma side is to reduce the first peak of the torque, but on the other hand the torque increases in the final part of the disruption. Moreover, the option with two straps has the advantage to strongly reduce the vertical torque. The toroidal torque values are almost independent on the different options. On the other hand, the two straps option has the disadvantage to produce a slightly bigger net force applied to each blanket.

References:

[1] M. Sugihara, M. Shimada, H. Fujieda, et al., Disruption scenarios, their mitigation and operation window in ITER, Nuclear Fusion 47 (2007) 337–352.

P1.72 (Poster/Topic G : (In)-Vessel Engineering and Remote Handling): Mon, 16:00–18:00 Foyer

Gamma dose distribution optimization in devices for testing facilities of remote handling systems for fusion technology: Methodology and results

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Future fusion reactors will require remote handling systems due to their neutronic activation and subsequent gamma irradiation inside the chamber. The testing and validation of these systems will be carried out in facilities specifically designed for this purpose. The aim of this paper is to describe a methodology to optimize both a bremsstrahlung generated gamma dose and its spatial distribution inside a given testing volume. Electron main beam spectrum and intensity, angular distribution of the split beams and target material and its thickness are the main considered parameters. Dose distribution at any given point of the testing volume is then obtained in order to perform a statistical analysis which establishes a criterion to choose the most suitable parameter configuration for the different irradiation needs.

P1.73 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

A one-dimensional flow process diagram model for dynamic and transient tritium transfers between ITER HCLL TBM auxiliary systems

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Demonstration of tritium self-sufficiency and tritium control are major scrutiny issues of fusion nuclear technology. Tritium transfers flow process diagrams (FPD) are demanded in order to quantify tritium transfers between breeding module and auxiliary systems in the TBM internal breeding cycle in ITER. FPD models appear as key for the scaling and design of tritium processing systems (coolant purification systems – CPS, tritium extraction systems – TES).

The pulsed nature of ITER machine and the complex flux couplings between different systems demand inherently transient and dynamic modelling tools for transfers.

A 1-dimensional flow process diagram model for dynamic and transient tritium transfers between ITER HCLL TBM and auxiliary systems is proposed. The model is implemented by the use of TMAP7 tool, an ITER QA validated tool for tritium transport analyses.

Although being 1-dimensional, a major value of the model is its physical reliability on tritium transport assessments, in particular in the dynamic coupling of primary coolant chemistry (oxidations and/or H_2 swamping) and Eurofer evolution determining permeation fluxes. This aspect appears key for an ultimate choice and design of auxiliary systems.

The 1D FPD enclosure model implementation in TMAP7 is described. Set of parametric representative conditions are given for external H_2/H_2O control additions. For nominal TBM/HCLL systems design thermal-hydraulic inputs, an optimum choice of primary coolant chemistry parameters is proposed and argued in terms of processing systems design characteristics.

P1.74 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

Thermo-mechanical experiment and analysis on an HCPB-TBM mock-up

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In the frame of the R&D activities on the Helium Cooled Pebble Bed Test Blanket Module (HCPB-TBM) to be tested and qualified in ITER, ENEA CR Brasimone and the Department of Nuclear Engineering of the University of Palermo (DIN) have performed the test campaign and the benchmark activities on HEXCALIBER mock-up.

HEXCALIBER is a medium scale mock-up that represents a portion of the HCPB module with two lithium orthosilicate pebble bed cells and two beryllium pebble bed cells both heated by flat electrical heaters. This mock-up has been tested at the HE-FUS 3 facility of ENEA CR Brasimone under adequate adjustment of bed temperatures, temperature gradients, coolant temperatures, flow distributions and mechanical constraints. The thermo-mechanical behaviour of both beryllium and lithiated ceramics pebble beds has been investigated under steady state and cyclic power conditions.

This paper describes the main results of both the experimental tests and the validation of the pebble bed thermo-mechanical constitutive model developed at DIN and implemented in the ABAQUS code.

P1.75 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

The European breeding blanket test facility: An integrated device to test European helium cooled TBMs in view of ITER

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The activities of testing and qualification of the two European TBMs to be installed in ITER, as well as their main ancillary circuits, are of outstanding importance in view of the experimental activities planned during ITER lifetime. Having in mind this purpose the EBBTF has been designed and installed at ENEA Brasimone Research Center in order to meet the important requirements coming directly from the scientific communities to tests and qualify blanket components and technologies in view of ITER and DEMO reactors. ENEA is responsible for the activities of design, construction and installation of the system at ENEA Brasimone and for all of them different national companies have been involved. Essentially, EBBTF consists of the upgraded HeFus3 helium loop coupled with a new lead lithium loop named Integrated European Lead Lithium Loop (IELLLO). When ready to be operated EBBTF will have unique characteristics in terms of size, power and flexibility, making possible to test: HCLL-TBM mock-up, up to 1:1 scale, HCPB-TBM mock-ups, up to scale 1:1 and technologies and components relevant for both TBMs as well as DEMO reactor blanket concepts and main TBM ancillary circuits. The device and its original design solution are described in this paper.

P1.76 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

Tritium permeation into and recovery from He coolant for the two EU TBM concepts

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A challenging issue for the European test blanket modules (TBMs) to be tested in ITER is the ability to measure and to remove the fraction of tritium permeated from the breeding region into the helium cooling system (HCS), keeping a suitable composition of impurities in the coolant. The need to remove the amount of tritium permeated into HCS from the breeder comes from the requirement to keep low the tritium partial pressure in HCS, thus controlling the release of tritium into the working environment via permeation or leakage. On the other hand, the possibility to remove, recover and account for tritium permeated is mandatory to keep controlled the TBM tritium balance and to correctly determine the TBM tritium breeding performance.

Although the very small tritium generation rate in all TBMs, with the consequent low tritium permeation rate into HCS, the experimental activity in ITER on this issue is important because of the possibility to measure the effective tritium permeability through the cooling plate system in a realistic fusion environment, a parameter which is of basic significance for DEMO. Moreover, the experimental qualification of a coolant purification system (CPS) based on the so far foreseen technology will be possibly carried out, being also this an important achievement in view of DEMO.

In this context, the paper here proposed is focused on the development of a numerical model able to give a reliable estimation of the tritium permeation rate into HCS for helium cooled lithium lead (HCLL) and helium cooled pebble bed (HCPB) TBMs in an irradiation scenario ITER relevant, taking also into account the He chemistry effects. Such estimation is the basis to prepare a precise input specification for the design of the CPS, avoiding an excessive over sizing of this system and, consequently, optimising its space needs in ITER facilities.

Therefore, on the basis of the estimation of tritium permeation rate into HCS, the sizing of CPS for both EU TBMs will be provided, giving also a detailed indication of the needed instrumentation for an effective control and monitoring of the TBM He coolant conditions.

P1.77 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

ENEA study on vertical module segmentation for a DCLL blanket for DEMO

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In the frame of the EFDA studies aimed to development of a breeding blanket for the DEMO, ENEA studied a blanket using both He and Pb-17Li as coolant (Dual Coolant Lithium Lead (DCLL)). The proposed solution for this blanket has been a Vertical Module Segmentation (VMS), relying on poloidal modules extending for the whole blanket length, both inboard and outboard of the plasma. These modules have a typical length of 10 m and a typical width of 1.5 m, using 32 inboard modules and 48 outboard modules. The modules can be extracted from the vertical upper ports in vacuum vessel and the cooling and breeding fluids can be feed both from vertical ports an from the lower horizontal ports.

The studies performed in ENEA and in University of Thessaly, Greece, showed that no problems arise in cooling, pressure drops, manufacturing and in MHD issues for a DCLL blanket VMS. The plasma disruption related loads are low in comparison with other solutions and result in relatively simple support systems. The temperatures are within the allowable limits for the reference materials. The path of the Pb-17Li in the modules is a straight path from lower to upper side of the modules, without changes of direction in the magnetic field of the tokamak. In order to reduce the MHD effects the structural grids of the module are separated from the flowing Pb-17Li by silicon carbide composite inserts. These inserts allow for a 2 mm gap filled with stationary Pb17-Li. The path of He is in two or three parallel cooling loops in grids, connected by intermediate manifolds to main He manifolds.

The advantages in this solutions are lower mechanical loads in case of plasma disruption, a very compact structure and a reduced path for Pb-17Li, leading therefore to lower MHD power losses.

P1.78 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

Design optimisation and measuring techniques for the neutronics experiment on a HCLL-TBM mock-up

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A neutronics experiment is in preparation in the frame of the European Fusion Technology Program on a mock-up of the EU test blanket module (TBM), helium cooled lithium lead (HCLL) concept, with the objective to validate the capability of the neutronics codes and nuclear data to predict nuclear responses, such as the tritium production rate (TPR), with qualified uncertainties. This is a follow-up experiment to a previous similar one conducted on the helium cooled pebble bed (HCPB) concept. As in the previous case, the TPR will be measured using Li_2CO_3 pellets (containing both natural and ^{6}Li-enriched lithium) located at various depths, at two symmetrical positions at each depth. Three independent measurements of the TPR will be performed by ENEA, TUD and JAEA.

Other measurement techniques are being developed and will also be used to measure the TPR, such as thermo luminescence detectors (TLDs), through distinct measurements of both the absorbed dose due to the energy released in the (n,t) reaction and in the decay of tritium. Diamond detectors covered with ^{6}LiF will be also used. The T and alpha products from the ^{6}Li(n,a)T reaction are detected and resolved due to the excellent energy resolution of the diamond detector and thus converted into ^{6}Li reaction rate via calibration in a thermal flux. The neutron flux in the LiPb will be measured using the activation foil technique, and the neutron flux spectrum will be measured as well down to thermal energies, relevant for TPR, using the time-of-arrival technique with a 3He counter. The measured quantities (E) will be compared with the same calculated quantity (C). C/E ratios will be provided, together with the related uncertainties.

A pre-analysis of the experiment has been carried out in order to optimize the mock-up configuration so that the neutron spectra inside the mock-up are as similar as possible to those in the TBM in ITER. Moreover, sensitivity and uncertainty assessments have also been performed, to evaluate the calculation uncertainty due to the uncertainties of the neutron cross sections. The results show that the uncertainty of the calculated TPR due to cross sections uncertainties amounts to about 3–7% depending on position, the reactions (n,2n) and (n,3n) on Pb being to cause the highest uncertainty.

The paper presents the pre-analyses conducted to design and optimise the mock-up configuration for the experiment, including sensitivity/uncertainty assessments of the TPR, as well as the development work on the measurement techniques and their relevance for tritium and neutronics measurements in TBM tests in ITER.

P1.80 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

Experimental MHD-flow analyses using a mock up of a test blanket module for ITER

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A promising candidate for a breeding blanket in future fusion reactors is the helium cooled lead lithium (HCLL) blanket that is foreseen as a test blanket module in ITER to demonstrate its reliable performance with respect to heat transfer and tritium breeding capabilities. In this type of blanket the heat is entirely removed by helium flowing in small channels within cooling and stiffening plates. The liquid breeder, PbLi, moves only slowly through the breeder units to allow tritium removal and purification in external facilities. The flow of the electrically conducting breeder under the influence of the strong magnetic field confining the fusion plasma leads to magnetohydrodynamic (MHD) phenomena such as induced electric currents, higher pressure drop and different velocity profiles compared with hydrodynamic flows.

In support to the design of an ITER test blanket module a scaled MHD mock-up of a HCLL blanket module with several breeder units and manifolds has been built according to an original design concept developed at CEA. The mock-up has been inserted into the liquid metal loop of the MEKKA laboratory at the Forschungszentrum Karlsruhe and MHD experiments are currently being performed. Results for pressure drop in breeder units and manifolds, and electric surface potential distributions are presented for several liquid metal flow rates and different strengths of the applied uniform magnetic field. These results serve as validation data for numerical tools and provide the necessary input for relevant scaling laws required during the design phase of the ITER TBM.

P1.81 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

Influence of helium cooling channels on liquid metal magnetohydrodynamic flows in the HCLL blanket

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One of the blanket concepts proposed to be tested in ITER as part of the test blanket module program of the European Union is the helium cooled lead lithium (HCLL) blanket design. In this configuration steel boxes, called breeder units, are arranged in an array, separated by a stiffening grid, to form blanket modules. The deposited thermal energy is removed by helium flowing at high pressure and speed in channels integrated both in the walls and in cooling plates immersed in the liquid metal. These latter subdivide the breeder units into slender ducts where the lead lithium circulates under the influence of the strong plasma confining magnetic field. This gives rise to magnetohydrodynamic (MHD) phenomena whose effects on flow distribution have to be investigated to evaluate the performance of the proposed design. The established MHD flow is affected by the presence of helium channels in cooling and stiffening plates that results in non-homogeneous wall conductance.

In support to the conceptual study of a HCLL blanket, numerical investigations of fully developed MHD flows in breeder units have been performed, taking into account both the presence of helium channels inside stiffening and cooling plates, and the multi-channel effects due to the exchange of currents through common separating walls. It has been observed that when helium channels are located in walls perpendicular to the magnetic field internal parallel layers spread into the fluid yielding a non-uniform velocity distribution in the ducts. Moreover, when the cooling channels are present in the walls aligned with the magnetic field, velocity profiles in the parallel boundary layers show a "wavy" behavior along field lines. Results have been compared with those obtained in the case in which cooling channels are omitted and the real thickness of the walls is scaled depending on the volume fraction of helium in the considered plate. This approach, based on the definition of an effective wall thickness, has been used in previous numerical studies as well as in experiments.

P1.82 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

Measurements of time-dependent liquid metal magnetohydrodynamic flows in a flat rectangular duct

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As a European reference concept for a liquid metal breeding blanket to be tested in ITER the helium cooled lead lithium (HCLL) blanket has been chosen. In this design the heat is removed by helium cooled plates which are aligned with the strong toroidal magnetic field that confines the fusion plasma. The liquid breeder lead lithium circulates through gaps of rectangular cross section between the cooling plates with a minimum velocity that is required to transport the generated tritium towards external facilities where it is separated from the liquid metal.

Magnetohydrodynamic liquid metal flows in conducting rectangular ducts exhibit jet-like velocity profiles in the thin boundary layers near the side walls which are aligned with the magnetic field like the cooling plates in HCLL blankets. The velocity in these so-called side layers may exceed several times the mean velocity in the duct, and it is known that these layers become unstable for sufficiently high Reynolds numbers. The present paper summarizes experimental results for such unstable time dependent flows in strong magnetic fields, which have been obtained in the MEKKA liquid metal laboratory of the Forschungszentrum Karlsruhe. In particular, spatial and temporal scales of perturbation patterns are identified. The observed time dependent flows could result in increased transport properties compared to steady state laminar flows that should be taken into account in future analyses of corrosion and tritium permeation into helium cooling channels.

P1.83 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

Design update and thermo-mechanical analysis of the EU-HCPB TBM in vertical arrangement

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In the frame of the activities of the EU Breeder Blanket Programme and of the Test Blanket Working Group of ITER, the Helium Cooled Pebble Bed Test Blanket Module (HCPB TBM) system is developed in FZK to investigate DEMO relevant concepts for blanket modules. In ITER the three main functions of a blanket module (removing heat, breeding tritium and shielding sensitive components from radiation) will be tested in DEMO relevant conditions using a series of 4 TBMs, which are irradiated successively during different test campaigns. In each campaign one HCPB TBM will be installed into the vacuum vessel connected to one equatorial port. Due to the installation of a correction coil for compensation of ripple effect, the two TBMs located in each equatorial port will be vertically oriented. As the studies performed up to 2006 in FZK concerned a horizontal orientation of the HCPB TBM, a global review of the design was necessary to match with the new ITER specifications. The new vertical design of the HCPB TBM, as well as the thermo mechanical analysis performed for the validation of the design, are presented in this paper.

Even if the design of the vertical HCPB TBM is mainly based on the architecture derived from the horizontal TBM, the change of configuration has significant impact on the design of the TBM subcomponents, e.g. TBM first wall, caps, stiffening grids and manifold system. The robust HCPB box (first wall and caps) is reinforced by an internal structure of stiffening grids and is able to withstand the full pressure of coolant helium (8 MPa) in case of in box loss of coolant accident. The cooling channels arrangements of the first wall, caps and stiffening grids have been adapted to the new outer dimensions of the TBM sub-components. In addition the helium flow scheme has been updated, rearranging the manifold structure in the back plate collector (a fourth manifold is added). The fluid dynamic and thermo hydraulic analyses performed justify the choices for the new flow scheme. The Breeder Units (BU) modular arrangement inside the box has been changed into 2 (toroidal direction) $\times$ 8 (poloidal direction) BUs to cope with the restricted space requirements in vertical arrangement. Considering the heat loads resulting from the updated neutronic analysis, together with the other elementary loads, a global thermo mechanical analysis of the vertical HCPB design has been performed, with the aim of verifying the accordance of the mechanical behaviour with the criteria of SDC-IC. The global assessment of the updated design takes into account the capability of the vertical TBM to cope with experimental needs (e.g. with regard to neutronic and EM behaviour).

P1.84 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

Impact on the EU-HCPB-TBM design due to change of orientation from horizontal to vertical arrangement

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In the frame of the activities of the EU Breeder Blanket Program and of the test blanket working group, the FZK develops the helium cooled pebble bed test blanket module (HCPB-TBM) for ITER. The goal of the TBM project is the experimental testing of the HCPB-TBM concept with regard to DEMO relevant issues (e.g. tritium breeding performance, irradiation behaviour and the capability to withstand electro-magnetic loads) in the frame of four test campaigns in the ITER schedule dedicated to different plasma scenarios and pulse conditions.

In ITER three equatorial ports are dedicated to TBMs where each of the ports is closed with one port plug. Every port plug contains two TBMs of different concepts which are tested simultaneously, so six TBMs are tested in parallel during operation of ITER.

The TBMs structural components are mainly manufactured of low activation ferromagnetic steel, in case of the EU-HCPB Eurofer. According to analysis it turned out that the magnitude of distortion of the plasma due to the mass of ferromagnetic material of the TBMs installed in between the toroidal coils of ITER is unacceptable. Therefore the installation of correction coils in the radial/poloidal plane in the middle of the port plugs plasma facing side is mandatory to compensate the distortion. Thus, a vertical separation of the port plug frame is required, resulting in a vertical orientation of the TBMs with the correction coil being installed in between.

This paper describes the impact on the HCPB-TBM design due to the change of the orientation of the EU-HCPB-TBM from horizontal (1 208 mm toroidal $\times$ 740 mm poloidal for TBM 2007) into vertical configuration (484 mm toroidal $\times$ 1 660 poloidal). The general modular design concept of the TBM based on the TBM box with internal stiffening grid as well as the breeding unit (BU) design remains. But due to the geometric requirements of the port plug frame the BU orientation inside of the TBM box has to be changed from 6$\times$3 BUs (horizontal$\times$vertical) to 2$\times$8 BUs. Therefore the sub-components of the TBM box (especially the first wall and the caps) require complete re-dimensioning with regard to structural- and thermo-hydraulic design. In the frame of the design update the manifold system of the TBM for coolant distribution between the different TBM sub-components is under review. Also a new concept for the attachment system for connection of the TBM to the port plug shield will be integrated into the new TBM version 2008-1. Additionally the design of the TBM will be checked from fabrication point of view with regard to fabrication aspects resulting from the re-orientation of the TBM.

P1.85 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

Experimental investigation on the possible techniques of pebbles packing for the HCPB test blanket module

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The helium cooled pebble bed (HCPB) test blanket module (TBM) features an array of breeder units (BUs), which are enclosed by the first wall, the side walls, the caps and the manifold back plate of the TBM. The manufacturing of the TBM involves several welding processes and due to manufacturing and materials technology reasons post-welding heat treatment (PWHT) of the TBM is highly recommended. In order to test the tritium breeding capability of the HCPB TBM, the BUs will be filled with pebbles of the solid breeder (lithium ceramic) and the neutron multiplier (beryllium). To avoid any chemical compatibility problems and/or exceeding the temperature limits of the involved materials under the high temperatures occurring during the PWHT, the BUs may be filled with pebbles after the TBM is built and heat treated. Therefore it is required to experimentally demonstrate that the BUs can be packed with pebbles through small filling holes (5 mm in diameter) in the caps of the TBM. The objective of this experimental study is to present an attractive technique for packing the BU with pebbles under the TBM-relevant manufacturing conditions. To pursuit this objective, the following tasks were performed:

1. Investigating different layouts (location and number) of the filling holes,
2. finding the minimum number of filling holes required for acceptable packing, and
3. utilizing the suction force, created by a difference in pressure, to transfer the pebbles from their container into the BUs and achieve dense packing.

In addition packing the middle BU, which is surrounded by other BUs, was experimentally demonstrated by using a small-diameter tube that passes through the cap and the stiffening grid down to the middle BU. A mock-up that simulates the HCPB-TBM-relevant BU was built from Plexiglas to conduct the required experiments. Spherical glass pebbles were used to simulate the beryllium pebbles. Mechanical vibrations (vertical and horizontal) and tilting of the mock-up were applied, during the packing process, in order to distribute the pebbles consistently in the whole interior volume of the mock-up and obtain dense packing.

P1.86 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

A multi-physics model for MHD flows in fusion relevant applications

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The numerical investigation of MHD flows in complex 3D geometries is of great relevance for the design and operability studies of the next step liquid-metal breeding blankets, which will be implemented in the forthcoming fusion experimental devices, like ITER and DEMO.

The analysis of such flows is complicated by the simultaneous presence of multiple physical phenomena: Fluid dynamics and heat transfer in the bulk of the fluid due to imposed pressure gradients, wall temperatures and internal volumetric heating, MHD effects in the fluid volume, where currents are induced by the presence of external magnetic fields, and electrostatics in the metallic enclosure walls, where the voltage induced at the fluid boundaries is applied to the neighbouring walls.

To account for all these different effects, which are mutually affecting each other, a complete analytical model has been developed, assembling a partial differential equation system to represent the physics involved behind the phenomena. This equation system was then implemented in the commercial software package COMSOL Multiphysics.

In the present paper, such analytical model is reviewed and critically commented, and some verification and reference cases of reactor-relevant configurations are analyzed using the software tool, to show the robustness of the analytical model and the suitability of the commercial code to handle complicated geometries, which are common in the design of fusion experimental devices.

P1.87 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

Thermal mechanical finite element model of an experimental mock-up simulating a solid breeding blanket

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A technical challenges for the successful development of fusion energy is the development and qualification of materials and numerical models for the breeding blanket components.

Tritium breeding materials, which have the ability to react with neutrons and produce tritium, are required to fill fusion reactions. Tritium is produced inside the breeding blanket by neutron capture in the lithium compounds (tritium breeders). The use of lithium compounds in form of ceramic pebbles packed in beds is a promising concept for blankets. Worldwide efforts have been dedicated to its R&D. One of the main issues, in the fusion engineering field, is to develop a numerical model that take into account all thermo-mechanical characteristics of pebble beds to simulate the breeding blanket complex geometries. This problem is very complex because it is necessary to simulate not only the thermo-mechanical characteristics of the material (conductivity, stiffness, mechanical resistance) but also their variations versus the temperature and the deformations. Presently an exhaustive theory of the thermal-mechanical behaviour of the pebble bed has not yet been developed. At the Pisa University a research activity is carried out on the breeding blanket of nuclear fusion reactors. In particular, standard tests to determine the conductivity of granular materials were performed. Moreover, in collaboration with UCLA, several tests on the creep behaviour of single ceramic pebble have been carried out. By means of these tests, the thermal-mechanical behaviour of the examined pebble beds was implemented inside a homogenous finite element model. The developed numerical model has been applied for simulating the experimental tests performed on breeding blanket mock up. These tests were performed by the ENEA (Brasimone Research Centre, Italy). In particular the HELICA mock-up has been tested to investigate the behaviour of ceramic pebble beds in reactor-relevant operating conditions, providing useful data. This paper shows the simulation of the HELICA experiment, especially, from the thermo-mechanical point of view using the finite element model of the pebble bed. The numerical results obtained have been compared with the experimental ones under the same geometry and loading conditions. The comparison shows a good agreement between the numerical model and the experimental results in terms of both temperature and thermal flux profile.

P1.88 (Poster/Topic H : Fuel Cycle and Breeding Blankets): Mon, 16:00–18:00 Foyer

A Monte Carlo study on the possible lay-out influence in the HCLL-TBM nuclear response

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Within the European Fusion Technology Programme the helium-cooled lithium lead (HCLL) breeding blanket concept is one of the two EU lines to be developed for a long term fusion reactor, in particular with the aim of manufacturing a test blanket module (TBM) to be irradiated in ITER equatorial port.

The Department of Nuclear Engineering of the University of Palermo (DIN) has been involved, for a long time now, in the study of the nuclear response of the liquid metal TBM.

In this framework, the TBMs main nuclear parameters, such as power deposition, tritium production and radiation damage, have been investigated in a previous work, taking into account its most recent design and a poloidal layout for both the module and its steel supporting frame.

As also a toroidal layout has been taken into account for the HCLL-TBM, at the DIN a research campaign has been performed to study its nuclear response in this lay-out with the aim to investigate the possible lay-out influence on the module nuclear behaviour. Such a study is the object of the present paper, that is focussed on the issues concerning nuclear power deposition, tritium production and radiation damage.

A computational approach based on the Monte Carlo method has been followed and a 3D heterogeneous model of the HCLL-TBM has been set-up, simulating realistically its new lay out and taking into account 9% Cr martensitic steel (Z 10 CDV Nb 9-1) as structural material. The analyses have been performed by means of the MCNP-4C code, running on a cluster of 16 workstations (64 cpus) through the implementation of the Parallel Virtual Machine software. A large number of histories have been simulated (2×10^9) for each analysis and the results obtained are affected by statistical uncertainties lower than 1%.

The main features of the HCLL-TBM nuclear response have been determined, such as power deposited together with the spatial distribution of its volumetric density, daily tritium production with the radial distribution of its volumetric density and structural material damage through DPA and He and H gas production rate.

The results obtained are reported and critically discussed.

P1.89 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Neutron irradiation effects on optical absorption of KU1 and KS4V quartz glasses

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Diagnostic and remote handling systems in fusion reactors will require the use of optical components (windows, lenses and optical fibres) which will be subjected to high levels of radiation during hundreds of hours. For remote handling applications the optical components will receive gamma radiation of the order of 10 Gy/s but in the case of diagnostic applications, in addition to a higher level of ionizing radiation the material will suffer neutron irradiation. Both types of radiation will affect the optical properties of the materials. Optical absorption is one of undesirable effects induced and it is therefore necessary to determine its change due to the two different types of radiation.

KU1 (high OH content, 800 ppm) and KS-4V (OH content < 0.2 ppm) quartz glasses, known as highly radiation resistant, are being examined within the ITER diagnostics program [1, 2]. In this work we have compared, for these materials, the effects of neutron irradiation (10^{21} and 10^{22} n/m^2 fluences) and gamma irradiation (23.8 MGy dose) on optical absorption from the UV to IR. Commercial Infrasil I301 (OH content < 8 ppm) has also been studied for comparison.

Important differences have been observed between gamma and neutron irradiation effects in fused silica: While gamma induced optical absorption depends on the material grade [3], the UV optical degradation of the silica after neutron irradiation at the highest fluence is similar for the three grades studied.

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P1.90 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Electrical and optical surface degradation of silica due to superficial He implantation

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Silica will be used in ITER and beyond in optical components and also potentially as an electrical insulator in different heating and diagnostic systems. The material will be subjected not only to neutron and gamma irradiation, but also to particle bombardment, due mainly to ionization of the residual gas and acceleration of the resulting ions by local electric fields. Hence the insulators in a fusion reactor will be subjected to a wide energy spectrum of ion bombardment ranging from $\leq$ keV up to MeV. As a consequence of this hostile environment severe electrical and optical degradation is to be expected. Previous work has shown that the vacuum face of not only silica, but also other candidate insulators for fusion applications, rapidly degrades when subjected to bombardment by protons and alpha particles with energies from 27–900 keV. Such vacuum surface degradation was also observed to occur after 1.8 MeV electron irradiation. This type of surface degradation was shown to be due to loss of oxygen caused by radiolytic preferential sputtering.

The aim of the work presented here is to study this type of degradation for lower energies, where the ion implantation is even more superficial. KS-4V samples were implanted with He ions from 21 down to 5 keV at 13°C. In this way very narrow He implanted profiles were produced. The evolution of the surface electrical conductivity was measured in-situ for different doses. After implantation SEM X-ray analysis of the implanted surface was performed to check for oxygen loss, and optical absorption measurements were carried out to evaluate the optical degradation.

The results show that surface degradation occurs more rapidly for the lowest energy (5 keV) ions, indicating that the superficial narrow implanted He profile must play an important role in the surface degradation, the process being both accelerated and apparently enhanced by the presence of a high density of He near the surface.

P1.91 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Feasibility of a neutron diagnostic for the IFMIF test cell

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A diagnostic based on a set of fission ionization chambers is proposed for the IFMIF test cell in order to control on-line the irradiation parameters. Fission chambers are basically ion chambers with a fissionable material coating on the inner wall. The fission fragments provide a very large pulse from the neutron-induced reaction. These chambers are widely used in fission reactors; however the IFMIF environment (high neutron fluxes, gamma doses, reduced available space, etc.) requires a special design for these monitors and for the whole diagnostic. This work shows the current status of this diagnostic, which is being performed in the framework of the Broader Approach activities.

The definition of the diagnostic strategy is addressed, establishing the main specifications and requirements of the fission chambers and their deployment. A fast response time, a high signal to noise ratio and a long term reliability under intense neutron irradiation have been identified, among others, as main technical requirements. The electronics accompanying this complex system is another main issue, and also is discussed. The experimental system will be also capable of detecting the gamma-rays contribution which perturbs the neutron signal, being necessary its removal.

Recent calculations considering the IFMIF high flux test module neutron spectrum show that U-238 and U-235 are the best fissile material candidates to measure fast and thermal neutrons, respectively. Some preliminary thermal calculations have been also performed considering the high neutron fluxes and gamma doses of the possible locations, as the operational temperature of the diagnostic is a critical issue. In addition, the flux map obtained from the fission chambers signal has been assessed taking into account their sensitivity, their positioning inside the test cell and the IFMIF neutron flux. Results show that the experimental system will be capable of measuring the high neutron fluxes expected for IFMIF.

P1.92 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Tritium permeation experiment at IFMIF medium flux test module

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Tritium release experiments using different types of breeding material candidates are planned for the medium flux region of the IFMIF test cell. Nowadays, only ceramic breeder materials have been suggested to be tested in the tritium release module located in the medium flux test module of IFMIF. Liquid breeder blankets are very promising and for that reason, several concepts will be tested in ITER. One of the main problems concerning the liquid blankets is the permeation of the generated tritium in the breeder throughout the walls. Since tritium permeation is very influenced by the irradiation conditions, IFMIF is a suitable scenario to perform tritium permeation related experiments.

In this paper, a preliminary design of tritium permeation experiment for the medium flux test module of IFMIF is proposed in order to contribute to the progress of the liquid breeder blanket concept validation.

The conceptual design of the capsule in which the experiment will be performed is carried out, taking into consideration the experiment necessities and its implementation in the tritium release module. Several thermal hydraulic calculations have been performed to evaluate the thermal behaviour of the irradiation capsule.

Proposal of an improved design of IFMIF test cell components for enhanced handling and reliability

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An adequate design of components to be manipulated by remote handling is a key factor in the success of any activated facility, having a decisive impact on availability, safe and prompt maintenance, occupational exposures and flexibility of the facility.

Such components should satisfy at least the basic remote handling requirements of modularity, standardization and assembling adequacy. More specifically the use of manageable modules, staggered alignment guides to prevent jamming, adequate access and visibility, lifting and griping fixtures for remote handling, always keeping simplicity and economy.

Highly activated components in the IFMIF facility are found at the test cell, a shielded pit where stepped shielding plugs and elongated test modules containing the samples are accurately located. The present reference design of the IFMIF test cell shows some drawbacks, in particular the jamming tendency of the shielding plugs, slow and complex access at the back plate and difficult access to the bottom of the test cell.

This paper summarises several modifications aiming at improving, under such remote handling requirements, the present reference design of the test cell shielding plugs and aspects of the geometrical structure of the test cell.

P1.94 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

BA materials activities: Radiation induced electrical degradation of HP SiC

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Within the EU-Japan Broader Approach (BA) materials activities, EURATOM/CIEMAT fusion association will carry out work to characterize basic properties of silicon carbide materials (SiC and SiCf/SiC), considered as potential long term reduced activation structural materials, as well as possible flow channel inserts in Li-Pb tritium breeding blanket modules. Measurements planned include volume and surface electrical conductivity, and H diffusion, as well as the effects of H and He on microstructure and electrical conductivity.

Studies of silicon carbide based ceramic composites for applications in fusion have been carried out for more than ten years, and although understanding of the basic radiation damage processes as well as microstructural evolution has shown significant advances, considerable further work is required to fully understanding the physical mechanisms and varied irradiation effects for the different forms of SiC and SiCf/SiC. Clearly in order to improve the radiation behaviour of SiCf/SiC, one must fully understand the basic response of the SiC matrix material.

Work has begun on hot pressed (HP) SiC. Results, now available for radiation induced degradation of the electrical resistance, show an order of magnitude increase in the volume resistance for irradiation in high vacuum to 400 MGy at 450°C, while at the same time surface resistivity in contrast markedly decreases. This difference in surface and volume behaviour is being examined. At the same time work is underway on the effects of H and He bombardment of the surface, radiation enhanced H and He diffusion, and microstructural modification.

P1.95 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Design of a beam dump for the IFMIF-EVEDA accelerator

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The IFMIF-EVEDA accelerator will be a 9 MeV, 125 mA cw deuteron accelerator prototype for verifying the validity of the accelerator design for IFMIF. A beam stop will be used for the RFQ and DTL commissioning as well as for the EVEDA accelerator tests. Therefore, this component must be designed to stop 5 MeV and 9 MeV deuteron beams with a maximum power of 1.12 MW.

The first step of the design is the beam-facing material selection. The criteria used for this selection are low neutron production, low activation and good thermomechanical behavior.

Neutron and deuteron activation calculations have been performed for a wide set of materials. As a result, quantities such as specific activity, contact dose rate, emitted photons, decay heat and waste disposal ratings have been estimated.

A thermomechanical analysis with ANSYS has been performed for a few materials which show good behavior from the radiological point of view and different beam stop geometries (conical, slab). For the conical beam dump the shape has been optimized with the aim of minimizing the peak power density received by the material (which ranges between 180 and 250 W/cm^2). The influence of the size and divergence of the beam at the beam dump entrance on the peak power density values has been also studied.

In this paper the present status of the beam dump design will be presented.

P1.96 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Preliminary mapping of the expected radiation damage of the bayonet IFMIF back-plate

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The back-plate of the target assembly system of the International Fusion Materials Irradiation Facility (IFMIF) is the most heavily exposed component to the high neutron irradiation flux and therefore its expected lifetime is estimated to be less than 1 year. For this reason the reference target design was conceived with a replaceable back-plate.

In the "bayonet"concept solution proposed by ENEA the back-plate is replaced "in situ"using a remote handling device, without moving the overall target assembly. The neutron damage evaluation of the back plate components is necessary in order to perform a qualification of the bayonet option.

A preliminary estimation of the neutron irradiation damage, in terms of displacement rates dpa/y, of the tightening mechanisms region is necessary in support of the back plate design to evaluate the behaviour of the structural material and the non-seizure coating.

The neutron spectra in the back plate zones covered by non-seizure coatings were calculated via the MCNP-4C2 code with the McEnea neutron source model based on the measurements of neutron emission spectra produced in Li(d,n) reactions for a thick lithium target performed at the Cyclotron and Radioisotope Center (CYRIC), Tohoku University, Japan. The neutron transport calculations have been performed considering the IFMIF full performance operation phase consisting of two beams of 125 mA of 40 MeV deuterons. The total neutron intensity of 9.85×10^{16} n/s (corresponding to 6.31×10^{-1} neutrons/deuteron) was used.

The neutronic analysis required evaluated cross sections data above 20 MeV, since the d-Li source of 40 MeV deuteron energy produces neutrons with energies up to 55 MeV. For the isotopes of tungsten, which is a constituent of the non-seizure coatings, the cross sections of LA-150N library, containing data for neutron energies up to 150 MeV, were used, while for sulphur, another coating component, the evaluated data of ENDF60 library up to 20 MeV, contained in MCNP package, were used, due to lack of cross sections at higher neutron energies.

Displacement damage rates (dpa/y) of the zones of interest were calculated.

The damage rates (dpa/y) were calculated by folding the neutron spectra with the displacement damage cross sections of the W and S isotopes. The heat deposition and gas production were also calculated following the same procedure.

P1.97 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Lithium target design analysis and criteria for IFMIF design

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A newly up-to date design analysis of the IFMIF target lithium jet is presented here. First of all we propose and justify a constant or quasi-constant back-plate curvature throughout the whole evolution of the jet. Moreover we reject the possibility of having a straight part of the back-wall once the jet has been powered. Trivial reason for this be the fact that one does not do efforts to provide overpressure and then leave them out exactly when the whole amount of power was just put in. This first issue implicitly undergo the criterion on no-boiling for the lithium. The design criteria for IFMIF should be discussed explicitly and, in the opinion of the authors, they should be related, by design, with the operative conditions of the specific sub-systems. The philosophy that we propose is quite simple: Relating the design margins to the local physical stability conditions. In particular the authors consider not to be concerned with the highest temperature, nor even with the lowest boiling margin (Ts-Tx), but with the boiling ratio (Ts-Tx/Tx-Ti) which expresses a first evaluation of the attitude of the system to go locally over the margin in terms of its actual way of behaving in handing out the power. Moreover we consider via an inner criterion much like that one of mean free path (or Debye sphere) how big is the spatial margin between the temperature peak and the boundary-layer in terms of the mobility of the local vortexes (comparing them to the sources of uncertainty of the 'safe' power deposition). This is once more something like the optical size of a reactor in reactor Physics. This job requires a detailed analysis not only to be conducted with design codes or usual CFD. We suggest also experiments in order to follow the spread of the vortexes and to try to foresee the mobility of the turbulent kinetic energy in the layer. Our criterion was set using some 5 factor basing on the sum of the uncertainties due to beam focusing, waves and inner pressure fluctuations. Superficial waves are also a critical issue. This can't be considered to be dealt with statistical fluid dynamics. Physics is needed; and a first issue could be driven by similarity analysis applications. More in general we propose that all what has no prescription due design rules, most of it should undergo this 'criteria' approach worked out here. More samples of this, mostly related with the lithium target physics and modelling are given throughout the paper.

P1.98 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Explorative experiment of steel erosion/corrosion in high speed flowing lithium

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In the frame of the R&D activities aimed to design the International Fusion Materials Irradiation Facility (IFMIF), the necessity arose to evaluate the corrosion/erosion behaviour of austenitic and ferritic/martensitic steels in flowing lithium at high velocity.

The IFMIF facility is aimed at the production of high flux (1 018 n/m^2/s) of 14 MeV neutrons to test the candidate Fusion materials under significant neutron damage, up to 50 dpa/year. The neutron production is obtained by a nuclear stripping reaction activated by 40 MeV accelerated deuterons impinging above a flowing lithium layer.

The intense power deposition (20 MW) in the lithium target volume corresponding to the footprint surface (200×50 mm), requires an elevate lithium velocity (from 15 to 20 m/s) to prevent any boiling phenomenon in the Li bulk and to assure an efficient cooling of the back-plate.

The high lithium speed is expected to provoke a significant erosion/corrosion effect on the metallic walls of the target assembly which are in direct contact with flowing fluid. This effect, if too much intense, would have, as widely demonstrated by hydraulic experiments, detrimental consequences on the stability of the hydraulic regime of the free surface flow and in turn on the correct power deposition of the back-plate.

The Lifus 3 facility in ENEA-Brasimone was built to deal with erosion/corrosion issue. A first 1 000 hours duration experiment in stable conditions of lithium velocity and temperature, was performed. The major results of the experiment were as follows:

- Preliminary erosion/corrosion rates of AISI 316 and Eurofer 97,
- quantification of the additional erosion/corrosion due to inlet/outlet flow effect,
- quantification of the additional effect of erosion with respect to simple Li corrosion,
- extrapolation of erosion/corrosion rate at higher Li velocities.

P1.99 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Approach to the lifetime assessment of the bayonet back plate for IFMIF target

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The IFMIF facility is aimed at the production of high flux (10^{18} n/m^2/s) of 14 MeV neutrons to test the candidate fusion materials under significant neutron damage, up to 50 dpa/year. The conceptual configuration of the IFMIF target, based on the bayonet back plate (BP), has been developed in the past years by several authors. The appropriate engineering design of the back plate, to be developed in the Engineering Validation and Engineering Design Activities (EVEDA) phase, would require a very high level of knowledge on the materials behavior under irradiation, that will be acquired only after some years of IFMIF experimental activities. For this reason the back plate which is primarily invested by the highest IFMIF neutron flux, has to be considered a sacrificial component. In spite of its systematic replacement, the engineering design has to be optimized and the life time analysis has to be made carefully, in order to credibly estimate the expected replacement frequency. Since the replacement time interval must be conservatively shorter than the BP lifetime and, at each replacement, the facility has to be stopped for, at least, one week and subjected to risky and uncomfortable operations, it is necessary to perform a trustworthy analysis of the lifetime. To this purpose the various interconnections between the main damaging causes are discussed in order to evidence the most plausible reasons of back plate malfunctioning. Due to the lack of knowledge in some fields and the early stage of design, the analysis is only semi-quantitative. The analysis, which accounts for erosion/corrosion, hydraulic stability, neutron damage and thermo-mechanical stress as the main damaging causes, evidences also the research areas which deserve foremost attention during the EVEDA phase. The considered malfunctions are: Lithium boiling, burning/piercing of the back plate, non-sufficient neutron flux, brittle rupture of the back plate, creep rupture, loss of tightness of the back plate sealing.

P1.100 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Experimental study of efficiency of natural oxide acting as tritium permeation barriers on Eurofer 97

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Tritium permeation from the breeder to the helium coolant is a fundamental safety issue in the design of the Helium Cooled Lithium Lead (HCLL) blanket system. The permeation of hydrogen isotopes through Eurofer 97 in different conditions was deeply studied in the past, demonstrating that it is necessary to reduce the permeated flux using tritium permeation barriers. A possible solution was identified in the nucleation and growth of natural oxides on the helium side of cooling system components. The major objective of this work was the experimental evaluation of the Permeation Reduction Factor (PRF) of natural oxides on Eurofer 97 steel. The growth of natural oxides was obtained adding a known content of water and hydrogen to argon, used in substitution of helium, flowing on Eurofer 97 surface. The oxide layer was produced in situ, during the permeation experiment. The PRF was measured on disk shaped specimens using the PERI 2 apparatus, at a temperature of 550°C. It was clearly demonstrated that, in a well defined range of water/hydrogen mixtures, it is possible to obtain a reduction of the permeated hydrogen flux of more than one order of magnitude. The experimental set-up and the obtained results are presented and discussed.

P1.101 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Proposal for a lithium heating experiment

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The IFMIF 40 MeV, 10 MW Deuteron accelerators can be assumed as CW source (176 MHz, 1 ns pulses) so that the transient heating leads to a maximum temperature increase rate of 6**E4 K/s at a lithium depth of 2.1 cm, where the Bragg peak is located. Bubbles, or any vapour development inside the lithium, would result in a Deuteron stopping range lengthen, with the risk of damages due to excessive thermal load in the wall. It seems to be highly recommended to plan a first lithium heating experimental activity in order to start to understand better problems that may face in the IFMIF facility. It has to be noted that such a test is not foreseen within the IFMIF-EVEDA phase; therefore the present study deals only with the technical feasibility without any implication in the present ENEA budget. In this framework the aims of a possible experiment, dedicated to the IFMIF case, may be summarized as follow:

1. To demonstrate lithium flow stability under beam-like thermal load for a representative sub-size experiment;
2. To observe the possible bubbles generation, their development and their effects (Pitting);
3. To study the effects of a reduced lithium speed with respect to the nominal value of 20 m/s using an electron beam, instead of a deuteron one, as heating source the deposition profile heavily changes.

In this paper detailed calculations of the electron heating deposition scenarios in a Lithium Heating Experiment (Lithex) and a detailed experimental configuration is proposed; the main aim is to evaluate the significance of a lithium electron heating side experiment in view of the IFMIF facility realization.

P1.102 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Multipurpose ANSYS FE procedure for welding processes simulation

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ANSYS FE procedures and 3D models for thermal and mechanical simulation of both Laser and TIG welding processes are presented. The special features are the applicability to a non uniform gap and the use of a fast iterative procedure that assures the constancy of the fixed max temperature along the single pass and between each pass and the following, apart from their shapes and sizes. All the thermal and mechanical material properties of both INCONEL 625 and AISI 316 are described as far as metallic vapour; convection and radiation effects are considered. The 3D ANSYS models use both brick and non linear elements and elastic and elasto-plastic materials. Several sensitivity analyses were performed changing mesh size, welding speeds and material properties: Results and comparisons are shown. A full simulation of a TIG welding specimen (source W7-X) with the root seam plus 14 passes is presented: Thermal and mechanical results after each pass with different elements sizes of the filling material were confronted. The mechanical calculation results very sensitive to the mesh shape: This fact implies very fine and regular meshes. Elements size of about 1.3 mm was uses for both the calculations. The specimen is first restrained and then welded with the foreseen welding procedure; after that it is released and the final linear and angular shrinkages are calculated. The ANSYS birth and death procedure is used and the CPU time was strongly reduced. The time calculation of some hours per pass and per meter can be considered for practical purpose.

P1.103 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Cavitation sensor for the IFMIF/EVEDA experiment

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In the IFMIF/EVEDA (International Fusion Materials Irradiation Facility, Engineering Validation and Engineering Design Activities) experiment the high lithium flow speed and the presence of several devices for the flow control increases the risk of cavitation. This could procure large jet instability and disturbances in the beam-lithium interaction. Bubble re-implosion in the liquid bulk could promote material erosion. Cavitation risk has to be consequently avoided. To this purpose a new arrangement for a cavitation sensor system has been developed. The system is based on a piezoelectric accelerometer transducer with a resonant frequency of 38–40 kHz. A new electronic set-up has been realized for a commercial sensor able to acquire signals and analyze them. A full set of test has been performed using the tests plant at Enea-Brasimone where water jet shaped by a nozzle flows on a 20 cm large bed with a 1–2 cm water depth. Jet characteristics can be parametrically varied in order to study the cavitation onset and development. The work has been performed in view of operation for the cavitation system in the Japanese EVEDA lithium jet experiment.

P1.104 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Microstructural examination of advanced beryllium grades for fusion application

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At present beryllium was picked as plasma facing material for first wall of the ITER and also as neutron multiplier for the TBM ITER and for blanket of the DEMO. Beryllium is subjected to serious deterioration of physical-mechanical properties under neutron irradiation swearing in swelling, radiation embrittlement and lowing of thermal conductivity. Radiation-induced degradation of beryllium properties depends on initial microstructure characteristics to a significant degree. Optimization of grain size, beryllium oxide and another alloyed elements and impurities content, anisotropy degree in beryllium will allow the increasing for the radiation damage resistance.

TEM-examination of several beryllium grades by Brush Wellman production were carried out. They are differed in microstructural parameters, powder morphology, consolidation methods. Under examination there were used the FEI Technai 20 FEG microscope equiped with an electron energy loss (EEL) spectrometer and the Philips 30CM microscope with using of high-resolution TEM (HRTEM) images by means of fast Fourier transformation (FFT). The TEM-discs for viewing in electron microscope were prepared by electropolishing in the TENUPOL device.

Distribution, morfology and composition of the beryllium oxide particles, the boundary between the particle and matrix and the grain boundaries were investigated. The estimation forecast for the beryllium oxide particles influence to properties of the beryllium grade after high dose neutron irradiation was made. The dislocation structure and distribution of impurities in beryllium matrix were also investigated. The comparison of the beryllium grades in tendency to the best radiation damage resistance were carried out.

P1.105 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Diffusion weld study for ITER test blanket module fabrication

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According to the current design for the European Helium Cooled Pebble Bed (HCPB) ITER Test Blanket Module (TBM) there are basically six subcomponents which have to be fabricated and assembled: First wall, caps, stiffening grid, breeding units, back plates/manifolds, and attachment system. The main technologies needed for blanket fabrication is joining of parts - particularly production of plates with internal cooling channels - and applying suitable post weld heat treatments. Both steps together are the key technologies that determine the mechanical strength of the blanket, the ductile-to-brittle transition temperature (DBTT) which is important under neutron irradiation, and the potential for a compact design. While it is certain that the structural material will be Eurofer or another, comparable reduced activation steel, most joining technologies and/or procedures have still to be developed, adapted, or qualified, although substantial advancements have been already reported. The designated fabrication route for plates with inner cooling channels is diffusion or solid phase welding which is either performed in a hot isostatic press (HIP) or in an uniaxial hydraulic press, both after different specific joint preparations. However, for an efficient TBM fabrication the application of different milling processes would be unavoidable. Therefore, the influence of six different common milling procedures on the diffusion weld properties has been studied by instrumented Charpy tests after a one-step and two-step low pressure welding process. The according microstructures have also been examined. Furthermore, the effect of nine typical states of surface contamination on the weld interface properties has been investigated and characterized. It could be demonstrated that two-step HIP diffusion welding can eliminate unfavorable surface fabrication defects and, therefore, might allow for more efficient milling processes.

P1.106 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Production of titanium beryllides with fine-grained structure

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Within the frame of the European Helium Cooled Pebble Bed (HCPB) blanket development, a considerable effort is developed to the qualification of ceramic breeder and beryllium neutron multiplier pebble beds. As the tritium inventory of beryllium pebbles has a main impact on the attractiveness and safety of the entire HCPB blanket, a major goal of the materials development is to maximize the tritium gas release under operating conditions. Preliminary investigations revealed that beryllides like $Be_{12}Ti$ may be much more suitable as neutron multiplier in future fusion power plants compared to pure beryllium. Titanium beryllides promise faster tritium release, much smaller swelling and better compatibility with stainless steel. However, considerable work is still required to develop efficient production methods for beryllide pebbles. During the last 6 years the firm Goraieb Versuchstechnik (GVT), located in Forschungszentrum Karlsruhe (FZK), in collaboration with FZK scientists has carried out a number of manufacturing tests aimed to produce titanium beryllides with fine grains. Microstructural analyses revealed that fabricated specimens consist, mainly, of titanium beryllides and have fine-grained structure. The status of different fabrication routes with appropriate microstructural analyses is described in this paper.

P1.107 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Influence of production parameters on the properties of the 13Cr-1W-0.3Ti-0.3Y2O3

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For specific blanket and divertor applications in future fusion power reactors a replacement of presently considered reduced activation ferritic martensitic (RAFM) steels as structural material by suitable oxide dispersion strengthened (ODS) ferritic martensitic steels would allow a substantial increase of the operating temperature from -550°C to about 650°. Temperatures above 700°C in the He cooled modular divertor concept necessitate the use of ferritic RAF-ODS-steels, which are not limited by a phase transition. Therefore a 13Cr1W ferritic ODS steel is being developed. Establishing of an optimum processing route, as well as analyses and qualification of the produced alloys by SEM and analytical TEM are major parts of the work. 13Cr ferritic steel powder together with 0,3 wt-% Y_2O_3 and 0,3 wt-% Ti powder was ball milled under varying milling parameters using a dedicated attritor mill. The milled powders were filled in specially designed stainless steel cans, degassed at 400°C for four hours and finally sealed. After this preparing step the powders were consolidated in a hot isostatic press (HIP) device. Furthermore thermo-mechanical treatments (rolling and extrusion) were carried out on the hipped materials. Earlier analyzes of RAF-ODS-steels showed a hardness of $\sim$ 430 HV and an ultimate tensile strength of $\sim$ 1270 Mpa at room temperature. Transmission electron microscopic (TEM) investigations of initially argon milled RAF-ODS-materials demonstrated a bimodal grain size distribution within the material's microstructure as well as trapping of argon to the ODS-particles after the HIP process. Due to this phenomenon the milling atmosphere was altered to hydrogen during mechanical alloying. This work concentrates on the effects of that change, by comparing analyses of hydrogen-milled RAF-ODS-materials. The influence of the thermo-mechanical treatments applied to the RAF-ODS-steels, which already improved significantly the ductility and impact properties of the RAFM-steel ODS-Eurofer, will be described in detail for initially argon as well as hydrogen milled RAF-ODS materials.

P1.108 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Mechanical and microstructural properties of EB welded RAFM ODS-Eurofer steel

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For specific blanket and divertor applications in future fusion power reactors a replacement of presently considered reduced activation ferritic martensitic (RAFM) steels as structural material by suitable oxide dispersion strengthened (ODS) ferritic martensitic or ferritic steels would allow a substantial increase of the operating temperature from $\sim 550°C$ to about 650°C. In all cases appropriate joining technologies have to be developed. Diffusion welding techniques to perform similar and dissimilar joints have been studied successfully. Friction stir welding (FSW) has shown a good potential but application is limited due to geometrical restrictions and needs further development.

For the advanced helium-cooled modular divertor concept various joining techniques are required for joining the complex structural parts made of different materials. The electron beam welding process with its highly concentrated energy input has been investigated as a potential process to join divertor structures made of ODS Eurofer.

For this purpose, samples of ODS-Eurofer steel were welded using a PTR 150 kV/15 kW EB welding facility. Two different post-weld heat treatments (PWHT) were applied to investigate their influence on the mechanical and microstructural properties of the welded joints.

Tensile tests in the temperature range between RT and 500°C showed a degradation of tensile strength, which did not exceed the strength of the non-ODS steel Eurofer. Charpy impact tests exhibited a significant decrease in upper shelf energy as well as an unfavourable shift in ductile to brittle temperature to higher temperatures. Microstructural investigations by means of light microscopy, scanning electron microscopy, and transmission electron microscopy revealed that the observed deterioration of the mechanical properties is mainly caused by dissolution and coarsening of the strengthening Y_2O_3 particles.

P1.109 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Impact of D-Li source photons on the nuclear responses in the IFMIF test modules

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The International Fusion Material Irradiation Facility (IFMIF) will use a thick lithium target bombarded with 40 MeV deuterons as neutron source for the irradiation of fusion reactor materials. To enable accurate neutronics calculations, the Monte Carlo code McDeLicious had been previously developed as enhancement to the MCNP5 code. McDeLicious relies on the use of evaluated cross section data for describing the d + Li reactions in the lithium target, sampling the source neutrons and transporting them – as well as secondary photons produced in neutron induced reactions in the test materials – through the facility. This approach was intensively validated against available measured neutron emission cross sections and thick target neutron yields. McDeLicious is now the standard computational tool for nuclear analyses of IFMIF.

In addition to neutrons, the interaction of deuterons with the lithium nuclei will also produce primary gamma rays via Li(d,xγ) reactions. These photons will initiate additional nuclear responses in the tested materials due to photo-nuclear reactions.

To enable a full evaluation of the d-Li source, the McDeLicious code has been further enhanced by taking into account the sampling of primary photons based on the use of evaluated d-Li cross section data. The assessment of nuclear responses produced by primary gamma-rays, in comparison with the source neutrons, has shown that they will increase the nuclear heating in the materials near the d-Li source by up to dozens percents. The contribution of primary photons to the displacement damage and the gas production turns out to be not significant.

P1.110 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Diffusion weld experiments on ODS-Eurofer

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The helium cooled divertor concept of a future fusion power plant requires various techniques to join the components made from tungsten, ODS-Eurofer, and Eurofer material. ODS-Eurofer is the oxide dispersion strengthened (Y_2O_3) variant of Eurofer which is a reduced activation ferritic martensitic 9CrWV reference steel. The use of ODS-Eurofer steel allows the increase of the operation temperature to more than 650°C. Unfortunately the usual fusion weld techniques are not applicable since they lead to joints ODS-ODS joints with mechanical properties of Eurofer steel decreasing the operation temperature. The deterioration is mainly caused by dissolution, evaporation and coarsening of the strengthening Y_2O_3 particles. One promising alternative kind of joining technique could be a soft diffusion weld (DW) process avoiding agglomeration and evaporation of the ODS particles. Therefore an experimental program for the development of an ODS-Eurofer DW process has been launched. The current paper presents the evaluation of DW parameters and the supplementary experiments. Two homogeneous ODS-Eurofer DW experiments and only one heterogeneous DW experiment had been carried out caused by lack of material. The tensile experiments are indicating a high temperature tensile strength close to the base ODS material.

P1.111 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Prospective testing and desing validation program for the IFMIF high-flux test module within the EVEDA phase

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Within the Engineering Validation and Engineering Design Activities (EVEDA) for the IFMIF Project, in support of the design process, a series of experiments will be conducted, leading from engineering tools validation up to tests of a full scale HFTM mockup. The general philosophy is to provide early feedback to the design process with smaller experiments, while the large scale experiments will provide a validation of the final design, with the opportunity to test also the control strategy for the temperature regulation system for the specimen stacks.

This approach is sustained by the revised modular design of the HFTM, which follows a "numbering up" scheme, by arranging 24 irradiation rigs in 8 compartments, each compartment containing a group of three rigs. In contrast to the previous design, where a static manifolding system had to distribute the flow to the four irradiation rigs, as well as to the lateral reflectors, in the revised design, the flow rate through each compartment can be controlled individually by valves. The manifolding problem on the compartment/reflector level is thus obviated, and further enables to experimentally investigate the thermal-hydraulic issues in a single compartment under realistic conditions.

The testing programm is therefore sequenced as follows:

1. The HFTM-SR "single rig" experiment: A single rig is investigated inside a suitable casing, so the minichannels conducting the helium coolant gas are reproduced similar to the situation in the HFTM. Neighboring rigs are substituted with heated surfaces. The rig will be instrumented with deformation sensors, such as miniature high temperature strain gages, and with thermocouples. The measured temperature and deformation data of the experiment will serve mainly as a validation case for the coupled thermal-hydraulic and thermo-mechanical analysis procedure. The experiment will also yield first information on the fitness of the capsule-rig coupling concept, where concurrent demands of thermal insulation, stability, positioning accuracy and machineability must be adjusted.
2. The HFTM-SC "single compartment" experiment: The second step will be a compartment with three rigs, which represents already the 1:1 case of the HFTM concerning thermal-hydraulics. It will allow to investigate the manifolding of the gas flow to the 16 parallel minichannels between the rig surfaces. This experiment will already allow to check the appropriate settings for gas inlet conditions and massflow, as well as of the electrical compensation heaters, which will lead to homogenous temperature distribution inside the specimen stacks. Since a low number of rigs is tested, the capacity of the data acquisition system, which is dimensioned for the 1:1 instrumentation as forseen for IFMIF operation, will allow a very detailed instrumentation of the rigs.
3. The HFTM-FA "functional assembly" experiment: The last step in the HFTM experimental program in EVEDA will allow tests on the loading and deformation of the thin outer container, as well as simulation runs testing the temperature control strategies for different irradiation/operation scenarios.

This sequence of experiments can be started in early 2009, when the HELOKA-LP helium loop facility has been commissioned. The duration will be till the end of the EVEDA phase 2012/13.

P1.112 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Hydraulics and heat transfer in the IFMIF liquid lithium target: CFD calculations

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The experiments designed and performed to study the thermohydraulics characteristics of the lithium flow are still far from the real conditions of the liquid Li target of the International Fusion Materials Irradiation Facility (IFMIF). For this reason, Computational Fluid Dynamics (CFD) calculation turns out to be a good approximation to the real behavior of the Li flow. Therefore, the aim of the numerical simulation is to determine the flow field of the high-velocity isothermal lithium jet flowing along a hard concave surface with vacuum over its free surface and a great energy deposition. A three dimensional (3D) modelling of the IFMIF design Li target loop, made with the CFD commercial code FLUENT has been carried out. Characteristics of jet continuity, deformation of the velocity field and jet thickness are main issues to determine the stability of the free surface. To obtain uniform neutron field and dose rate for testing materials, thickness of the target flow is necessary to be uniform. The real IFMIF design target geometry is modelled by a structural mesh of 450 000 hexahedral cells. The energy deposition (40 MeV) caused by deuteron beam is simulated as an energy source term inside the volume of liquid affected. A proper user define function (UDF) has been programmed for this energy source and the turbulence of the flow is modelled using the Reynolds Stress Model (RSM). A surface-tracking technique applied to a fixed Eulerian mesh called Volume of Fluid (VOF) is used for this purpose. This model allows the free surface tracking. The jet velocity with an initial temperature of 250°C has been varied from a range of 10–20 m/s. The main numerical result achieved is the proper stability of the flow, at least, in the central zone where the beam interacts with the fluid. However, the evolution of the free surface all along the concave back shows a slight separation of the flow at the end of the concave wall. On the other hand, calculated temperatures, 300°C at the free surface and 500°C inside the fluid in contact with are still below the boiling point of lithium at low pressures.

P1.113 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Comparison of neutron and gamma irradiation effects on fused silica monitored by electron paramagnetic resonance

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Fused silica is a material of high interest in thermonuclear fusion technology due to their use in optical components such as windows, lenses and fibres, for remote handling and diagnostic systems. These components will be subjected to high doses of neutrons, charged particles and gamma- and X-rays. Therefore, a detailed knowledge on the effects of such different kinds of irradiation on fused silica is necessary.

In this work, we have employed the electron paramagnetic resonance (EPR) technique to monitor the concentration of defects like E' and oxygen-related defects (POR and NBOHC) [1] produced by high neutron irradiation fluences (10^{21} and 10^{22} n/m^2) or by gamma irradiations (up to 12 MGy).

The EPR spectra show very different defect production on neutron or gamma-irradiated samples. So, after gamma-irradiations up to the highest dose, a similar maximum concentration of E' and oxygen-related defects is reached, about 4×10^{17} spins/cm^3, which is a value nearly constant after doses around 4 MGy. Whereas, for neutron irradiations at the highest fluence (10^{22} n/m^2) oxygen-related defects (mainly POR) reach a much larger concentration (up to 8×10^{18} spins/cm^3), as well as E' defects (9×10^{18} spins/cm^3). Moreover, for an increase of neutron fluence from 10^{21}–10^{22} n/m^2, the E' and oxygen-related defect concentrations increase by about a factor of 10 and 4, respectively.

Finally, the thermal stability of these defects has also been monitored by EPR using thermal treatments in air. A main conclusion from these treatments is that for gamma-irradiated samples a lower treatment temperature (about 400°C) is required to annihilate most of the observed defects than for neutron-irradiated ones (about 600°C).

References:

[1] K. Moritani, I. Takagi and H. Moriyama, J. Nucl. Mat. 325 (2004) 169.

P1.114 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Hydrogen transport and trapping in GlidCop Al25 IG alloy

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GlidCop Al25 IG alloy is an oxide dispersion strengthened (DS) copper alloy that was under investigation for ITER high heat flux components like the divertor or the first wall, because of its high thermal conductivity, good mechanical properties and thermal stability up to almost the melting point and good swelling resistance. Conversely, there are critical issues because of the alloy low ductility and fracture toughness after irradiation. Together with mechanical and thermal properties and irradiation performances, the characterization of hydrogen isotope transport properties is compulsory in any fusion material because these properties affect important issues, such as the fuel economy, plasma stability and the radiological security of the fusion reactor. To the knowledge of the authors, hitherto, hydrogen isotope transport parameters have not been experimentally determined for any kind of DS-copper. The hydrogen interaction properties of permeability, diffusivity and Sieverts, constant in the GlidCop Al25 IG alloy have been experimentally evaluated in this work, using the gas evolution Permeation technique. The experimental range used in the study has been a temperature from 300–520°C and high purity hydrogen loading pressures from 10^3–1.5×10^5 Pa. In previous experiments with ODS-RAFM steels, the presence of nanostructured oxide particles was demonstrated to affect enormously the hydrogen isotope transport behaviour in comparison to the corresponding base material [1]. In the case of the DS-copper alloy the influence of the ultrafine Al_2O_3 particles on hydrogen transport has been studied in comparison to the base material copper and other copper alloys. Hydrogen trapping phenomenon has been detected in the lower temperature range, showing an exothermic absorption of hydrogen below 415°C and a decrease in the effective hydrogen diffusivity. The corresponding trapping parameters of trapping activation energy and effective density of traps have been evaluated.

References:

[1] G.A. Esteban, A. Peña, F. Legarda, et al., Fusion Eng. Des. 82 (2007) 2634–2640.

P1.115 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

3D CAD-MCNP methodology applied to neutronic irradiation effects analysis in fusion reactors

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A new methodology for 3D neutronic calculations suitable for complex and extensive geometries is described, which enables accurate neutronic calculations for systems characterised by high detail levels. First, CAD detailed modelling of the system is fully performed and processed through a CAD-MCNP interface in order to generate an MCNP geometry input file. Neutronic irradiation results are then achieved running MCNPX directly on this previously obtained output. This procedure can be applied to fusion reactor analysis in order to accomplish neutronic calculations along the blanket, both in inertial and magnetic confinement systems, focusing on neutronic irradiation figures and their subsequent effects on the device.

P1.116 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Developing the IFMIF RAM planning

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To prove that nuclear fusion power plant is viable being commercial Demostration Power Plant (DEMO) will be build. International Fusion Materials Irradiation Facility (IFMIF) is a project with the aim to test samples of candidate materials up to about a full lifetime of anticipated use in fusion energy reactors. The materials will be irradiated by a neutrons source that is formed by two lineal accelerators of 125 mA each which collides with a liquid lithium target.

To fulfill its goals a very high availability rate is needed, overall availability of 70% including scheduled outages. This means that the requirement is 80.7% of scheduled working time. In this way it allows large irradiation simulation periods at the reactor's core in a relatively short time.

The availability specifications in IFMIF project are very high taking into account it's an experimental facility. For this reason has started to develop a Reliability, Availability and Maintainability (RAM) program that interact with the design process and a RAM group for IFMIF has been proposed in order to assure the coherence in the calculations of IFMIF's availability and to take part in the design activities to improve the availability.

As a first step, a critical review of the previous RAM activities develop during the last 10 years in the framework of the IFMIF program is made. In this work we also analyze the software tools available for RAM analysis and it is proposed that RiskSpectrum should be used in the future in the IFMIF project.

It is an interactive tool for reliability and safety analysis and it allows a complete organization analysis and presentation of risk and reliability information. As a first validation example, the target facility has been modeled using this code in order to benchmark the previous RAM results calculated by Markov chains.

P1.117 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Influence of self-interstitial mobility on He-vacancy cluster nucleation and growth in nickel

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Austenic steels suffer from changes in mechanical properties under irradiation. In particular, in the presence of He, void swelling is observed. For the safe operation of future fusion reactors this is an important issue that must be addressed. There is a significant number of experiments in the literature devoted to elucidate the role of He and defect production in the nucleation and growth of voids in these materials. However, in order to obtain a basic understanding of the processes involved, and even more importantly, to be able to extrapolate these results to fusion conditions, predictive modeling is required. This modeling should be based in the most reliable tools, such as first principles density functional theory, whenever possible. But it also requires of long term evolution of defects to reach those time scales in the experiments. We have used results from molecular dynamics simulations on defect mobilities and binding energies to study, using kinetic Monte Carlo, the desorption of He from implanted Ni. These results have revealed good agreement with experimental measurements: Two desorption peaks at temperatures similar to the experimental ones and three peaks for higher doses. From this comparison the reactions responsible for these peaks can be identified. There are however small discrepancies with the experiments that could be due to the use of values for binding energies obtained from molecular dynamics simulations. Ab initio calculations will be performed to compare with some of these values.

The presence of impurities could affect the results observed in pure Ni. In particular, impurities could interact with self-interstitial atoms, which are highly mobile, effectively reducing their mobility. In this work we study the influence of the mobility of self-interstitials on He desorption. The nucleation of He-vacancy complexes is studied depending on the mobility of these self-interstitials in terms of He to vacancy content as well as concentration of these complexes. For these calculations new parallel kinetic Monte Carlo algorithms will be tested.

P1.118 (Poster/Topic I : Materials Technology): Mon, 16:00–18:00 Foyer

Influence of estimated beam asymmetries of IFMIF linac on the energy deposition on the lithium target

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The International Fusion Material Irradiation Facility (IFMIF) will provide an intense high-energy neutron field for high fluence irradiations of fusion power reactor candidate materials. Neutrons will be produced by two 125 mA beams of 40 MeV deuteron striking a thick target of flowing liquid lithium. In normal operation, the footprint of each beam must be rectangular, 20 cm horizontal$\times$5 cm vertical. The beam profile must be approximately uniform (5%) with sharp edges to produce a neutron field with the characteristics adequate for the irradiation of the samples.

Several simulations have been performed for the IFMIF accelerator, taking into account linac element errors (quadrupole displacements and rotations, gradient errors, etc). Both static errors (which can be detected and corrected with appropriate diagnostics) and dynamic errors (they are uncorrected) have been considered. These errors give rise to a beam shape at the Li target different from the nominal one. In this paper, the influence of linac element errors on the beam shape and, therefore, on the energy deposition on the target and on its temperature is studied. The effect of the possible combinations of errors is analyzed and the resulting range of lithium temperature variations due to linac errors is deduced.

P1.119 (Poster/Topic J : Power Plants, Safety, Environm., Soc.-Economics): Mon, 16:00–18:00 Foyer

Safety analysis of ITER failures and consequences during maintenance

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This study has been a first attempt at identifying potential worker overexposure situations during machine maintenance operations. The key findings obtained are as follows. Firstly, we have found no machine maintenance operations where the risk of worker overexposure is considered significantly large that immediate design attention is needed. Secondly, the most significant risk of worker overexposure is due to airborne releases of radioactivity from cooling water pipes and tubes that may not have been fully drained and dried, when they are cut, or inadvertently opened, by workers (frequency of pipe-cutting activities could be significantly high). Thirdly, the risk of overexposure from human error could also be significant. This varies from mistaking the machine sector, to mistaking the component to be maintained. This is analogous to working on a live electrical circuit, when it is believed to be dead (disconnected from the power source) because the worker has mistakenly selected the wrong circuit a look-alike one. Similarly, consider the situation of a worker mistakenly preparing to work on a cooling water circuit that is still at pressure and temperature, instead of the one that has been drained and dried. The more look-alike situations there are in the facility, the greater the probability of committing this type of error. Fourthly, when consideration is given to human error, we believe that the aggregation of different diagnostics in the same port enhances the probability of human error. At the moment, these risks cannot be quantified. The task of quantifying those risks in the future should be considered. Finally, the transport of activated in-vessel components, including components of plasma-heating and current-drive systems, in non-shielded casks, could carry with it a significant risk of worker overexposure. In the context of ALARA, this approach requires a specific study to justify its use.

P1.120 (Poster/Topic J : Power Plants, Safety, Environm., Soc.-Economics): Mon, 16:00–18:00 Foyer

Efficient helium cooling methods for nuclear fusion devices: Status and prospects

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Helium cooled reactors offer several important advantages as the helium is an inert and neutron transparent gas, compatible with high outlet temperatures and all structural and functional materials considered for plasma facing components. Among gaseous coolants helium offers a very high heat conductivity, reasonable heat capacity and high temperature stability, while high operational pressures are needed due to the density characteristics. High temperature compatibilities are especially important in regard to the use of tungsten in the divertor, which has an extremely high ductile to brittle transition temperature.

Due to the important advantages both European reference blanket concepts are completely helium cooled systems and in addition, the development of a helium cooled high temperature divertor is in progress. Even the third, alternative European blanket concept "Dual-coolant lead lithium"in regard to a DEMO fusion reactor relies on a helium-cooled structure. This means that the likelihood of the first generation of European power reactors (DEMO and beyond) being helium cooled systems is extremely high. On the other hand, during the power plant studies performed up to now it became evident, that the helium pumping power could become very high if standard cooling methods based on hydraulically smooth cooling channels would be applied and that a very high heat flux in the divertor can only be achieved when using sophisticated gas cooling methods.

Against this background major helium activities were launched at the Forschungszentrum Karlsruhe, including the design and construction of several helium test facilities applicable to perform various experiments from single effect studies up to full component tests for the qualification of test modules to be operated in ITER. In this paper the main focus is on the description of the current status of helium cooling method development and its application to the in-vessel components of future fusion power reactor. Starting from the description of the currently foreseen cooling methods for blanket and divertor structures the prospects of further optimizing the cooling systems are assessed based on a comprehensive review of gas cooling methods studied in different fields of technology. This includes passive enhancement techniques for convective heat transfer particularly artificial roughness elements formed either in the form of cavities on an initially smooth surface (dimples, grooves) or as a system of protruding intensifiers (ribs, pins, microscopic surface roughness). These devices improve heat transfer primarily by perturbation and better mixing of the near wall flow by means of formation of secondary flows, by intensification of turbulence production and in some cases by formation of coherent fluid motions. The effectiveness of these methods for the considered components is determined as far as possible by combining available empirical data with specific geometrical characteristics.

P1.121 (Poster/Topic J : Power Plants, Safety, Environm., Soc.-Economics): Mon, 16:00–18:00 Foyer

Attachment concept for the 'Multi Module Segments' in a pulsed DEMO reactor

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The pulsed thermonuclear demonstration reactor (DEMO) features challenging operational conditions such as high neutron fluxes, high temperatures, and significant thermo-mechanical stresses. These conditions do not require only a selection of advanced structural materials, but also the development of reliable means to assemble the in-vessel components together; allowing thermal expansions, disassembly, and maintenance in attractive scenarios. Over the course of DEMO lifetime, the materials are subjected to embrittlement by neutron irradiation, swelling, considerable thermo-mechanical fatigue and creep. Traditional joining methods may be rarely used in the harsh fusion environment to assemble different components. In addition any proposed layout should cope with the limited space available inside the vacuum vessel. The component position relative to the plasma, number and lengths of the pulses and degree of shielding have a significant impact on the dimensions, geometry, and type of material to be used. The objective of this study is to review the proposed attachment systems (developed within the latest European DEMO conceptual study) for the vertical segmentation concept called 'Multi Module Segments' (MMS), identify the most important physical and technical requirements for these systems, and suggest an optimized solution considering these needs.

P1.122 (Poster/Topic J : Power Plants, Safety, Environm., Soc.-Economics): Mon, 16:00–18:00 Foyer

Issues from activation/dose rates calculation for full W ITER machine and increased fluence

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ITER activation calculations are essential for the ITER safety assessment in the area of normal operation, decommissioning, waste assessment, accident analysis, and worker safety. The paper presents and discusses the results of the activation calculations for ITER to updates the previous ones included in the Generic Site Safety Report (GSSR). The new analysis is based on the most recent ITER design information (updated at September 2007) and it has been performed using the most recent, reliable and validated nuclear data and codes. The results in GSSR refer to an average first wall (FW) neutron fluence of 0.5 MWa/m^2 with beryllium FW protective layers (PLs).

In the present analysis three configurations have been considered:

1. Average FW neutron fluence of 1.0 MWa/m^2 with beryllium FW PLs;
2. average FW neutron fluence of 0.5 MWa/m^2 with tungsten FW PLs;
3. average FW neutron fluence of 1.0 MWa/m^2 with tungsten FW PLs.

The approach used considers the Scalenea-1 radiation transport sequence for obtaining the 175 groups neutron fluxes in all the materials/zones on the radial direction of the ITER equatorial mid-plane and the EASY code package for the material activation analysis. Results obtained with the EAF2005.1 and the EAF2007 neutron activation libraries are compared. A comparison with GSSR results is also presented and discussed. For tungsten PLs, calculations have been also performed using a 1-D MCNP approach in order to investigate the effect of the multi-group treatment of cross sections compared with a continuous energy treatment on deriving the self-shielding effect.

The activation parameters calculated with EASY2005.1 are, especially for steel materials and for tungsten, significantly higher than the corresponding ones obtained for GSSR calculations. This is particularly relevant for cooling times from few days up to about 50 years after shutdown. The difference compared to the previous GSSR results (when the EAF99 activation library was used) is largely due to the neutron activation library EAF2005.1 used for the 2007 calculations. This especially can be attributed to the (n,p) cross section data of Ni-58 producing Co-58. The difference practically disappears if the EASY2007 code package is used instead of EASY2005.1 (due to a correction of the above mentioned cross section data).

P1.123 (Poster/Topic J : Power Plants, Safety, Environm., Soc.-Economics): Mon, 16:00–18:00 Foyer

Loss of plasma control transients in ITER: A review with AINA safety code

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In this contribution, two ITER losses of plasma control events are investigated. The first is a sudden termination of fuelling and/or external heating. The second studies a hypothetical sudden improvement of energy and particles confinement time, which might take place after reaching a new confinement mode. For the second event, initial simulation parameters are scanned in order to find out which could lead to highest fusion power.

A whole bunch of analyses has been done with AINA code, a comprehensive hybrid code comprising a zero dimensional plasma dynamics and a radial and poloidal thermal analysis of in-vessel components. AINA is an evolution of SAFALY code, which was initially adopted to assess ITER EDA plasma abnormal events from the safety point of view, and is intended to the quantitative investigation of the safety of nuclear fusion reactors such as ITER.

In the first case for the 500 MW scenario, for "and"condition, plasma terminates after 9.7 seconds by edge energy collapse, for "or"condition, plasma terminates after 3.5 seconds by edge energy collapse for auxiliary heating cut-off, and terminates by locked mode disruption for fuelling cut-off.

In the second case for the 500 MW scenario, a scan of initial ion temperature, of initial fusion power and of density profile has been done, in order to find which confinement time drives to a high fusion power. It has been found that low initial ion temperatures, high initial fusion powers and high-pitched density profiles favour a higher maximum fusion power.

FEEL-UPC research group has been developing AINA code from more than three years. AINA code brings significant improvements in relation to SAFALY code, which was used to perform ITER GSSR LOPC analyses.