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Introduction

Forschungszentrum Jülich (FZJ) as a EURATOM Association coordinates its fusion research activities within the Nuclear Fusion Project (KFS). The programme is based on several institutes and is well embedded into the European fusion research structure, where FZJ is now focussing on the two topics "plasma-wall interactions" and "ITER technology". The major part of the Jülich research activities is located within the Institute of Energy and Climate Research (IEK). The former Institute for Plasma Physics (IPP, now IEK-4 Plasma Physics) has by far the largest share of scientific staff in physics and technology for fusion, operates the TEXTOR tokamak, performs scientific work on JET and DIII-D, supports the Wendelstein 7-X construction and takes up significant projects related to the ITER development. IEK-2 (Microstructure and Properties of Materials) operates the high heat flux test facilities JUDITH 1 and 2 which are installed inside a hot cell and in a controlled area which is licensed to operate with toxic and radiating materials; this group represents the materials science expertise within the Jülich fusion programme. The Central Technology Division (ZAT) provides engineering expertise and specialised workshop capacities. The Jülich Supercomputing Centre (JSC) operates various types of supercomputer systems, among which one device (HPC-FF) is dedicated exclusively to fusion research within EFDA.

The Association EURATOM-FZJ has very close contacts to the neighbouring EURATOM associations in Belgium and The Netherlands. In 1996 they together have founded the Trilateral Euregio Cluster (TEC) which provides a clustering of resources in order to perform a co-ordinated R&D programme, to operate or construct large facilities (TEXTOR, MAGNUM-PSI), to combine different kinds of expertise and to allow for the forming of a strong partnership as a consortium within the ITER construction phase. An updated TEC agreement with a strong emphasis on the topic "plasma-wall interactions" and the joint use of dedicated facilities in Jülich, Rijnhuizen (NL) and Mol (B) has been signed in 2010.

Co-operations beyond Europe are strongly supported by an IEA Implementing Agreement on "Plasma-Wall Interaction in TEXTOR" together with Japan, USA and Canada, which meanwhile also serves as a basis for the exchange of scientists to other devices than TEXTOR.
Objectives and incorporation into the research area

Fusion research at Forschungszentrum Jülich is to a large extent scientifically organised along topical groups, i.e. plasma-wall interaction, tokamak physics, diagnostics, theory and modelling, and technology. These groups use a variety of different experimental facilities. Among these the most important machine is JET where scientists from Jülich are strongly involved, in particular in the scientific exploitation and also in the technical preparation of the new ITER-like wall project as well as in experiments addressing ELM-mitigation. Other facilities outside Jülich with participation from FZJ are DIII-D, PISCES-B, ASDEX-Upgrade, TS, LHD and MAST.

IEK-4 Plasma Physics operates the TEXTOR tokamak as a local facility in Jülich \( (I_{p,\text{max}} = 0.8 \text{ MA}, B_{T,\text{max}} = 3.0 \text{ T}, R = 1.75 \text{ m}, a = 0.46 \text{ m}, \text{plasma volume } 7 \text{ m}^3, \text{circular cross section, toroidal graphite belt-limiting (pumped)}, 16 \text{ TF coils, pulse length 12 s; auxiliary heating power: NBI co 2 MW, NBI counter 2 MW, ICRH 4 MW and ECRH 1 MW).} \)

The Dynamic Ergodic Divertor (DED) on TEXTOR provides unique means for resonant magnetic perturbations: 16 helical in-vessel RMP coils with base modes of 12/4, 6/2, and 3/1, \( I_{\text{max}} = 15 \text{ kA as well as DC and rotating fields of up to 10 kHz. Based on these means the programme participates in ELM-mitigation studies (joint experiments) and in the investigation of power exhaust in helical divertor structures in preparation of long pulse and steady-state operation in stellarators.} \)

For Plasma-Wall Interaction (PWI) studies a powerful PWI test facility is available on TEXTOR: two air-lock systems to expose movable and easily exchangeable larger scale wall components with gas feed, external heating and active cooling under ITER-relevant parallel heat and particle flux densities. The system is equipped with a comprehensive in-situ set of PWI diagnostics.

The former test stand for NBI heating provides a 60 keV, 70 kW hydrogen/deuterium beam on samples of 15x10 cm\(^2\) from 10 ms to 15 s duration as a high heat flux test facility (MARION). Possible upgrades for the test of larger components and more flexibility inside the target chamber are anticipated.

In addition the programme is supported by a variety of smaller laboratory devices: a tandem accelerator device for the quantitative determination of surface material compositions (NRA, RBS), dedicated laboratory devices for in-situ PWI simulation and analysis (TOF-SIMS) and various devices for the plasma assisted preparation of fusion relevant layers and coatings, and a "mirror laboratory" for the characterisation and analysis of experiments with plasma facing optical mirrors in tokamaks.

The special expertise of IEK-4 in fusion technology is manifested by major engineering projects: concept development, design, construction and installation of the TEXTOR tokamak including various major upgrades and recently the design, layout, manufacturing and assembly of the superconducting bus-bar system for Wendelstein 7-X, design and procurement for a bulk tungsten plasma facing component for the new JET divertor and the design and procurement for the target station of the new experiment Magnum-PSI at FOM. IEK-4 Plasma Physics has
taken up substantial new projects for the development of ITER, based on special national funding. The task comprises R&D and design work for the CXRS diagnostic port plug system, the development of a new laser-based diagnostic system for Tritium retention, and the construction of a fast disruption mitigation valve.

IEK-2 operates the high heat flux test facilities JUDITH 1 and JUDITH 2. These electron beam facilities are capable to deliver ITER- and DEMO-relevant quasi-stationary heat fluxes with loaded areas of up to 50 x 50 cm$^2$ and transient thermal load tests on a millisecond time scale with energy densities in the MJ/m$^2$ range to simulate Edge Localized Modes, plasma disruptions, and vertical displacement events. A unique feature of this test equipment is the operation inside a hot cell which allows for the testing of neutron irradiated and toxic materials such as Beryllium and Tritium-containing samples.

ZAT is developing and manufacturing experimental devices and techniques for a wide range of scientific applications which are not available on the market. This central FZJ facility provides expertise in the fields of project engineering, joining and testing technology, and prototype manufacturing using special tools and technologies.

The Jülich Supercomputing Centre JSC operates a dual super computing system (both: general purpose and massive parallel architectures) and hosts the first dedicated European Supercomputer for Fusion HPC-FF (100 Teraflop/s), which started operation in 2009 under an EFDA Implementing Agreement. HPC-FF is embedded into the European theory and modelling activities, such as the EU-ITM task force, and it also serves as a training platform for the Petaflop Computer for ITER, as part of the Broader Approach agreement between Europe and Japan.

The Helmholtz Association's fusion activities are based on the European fusion research programme. The following Helmholtz Centres are involved: Max Planck Institute of Plasma Physics (IPP, Garching and Greifswald), Karlsruhe Institute of Technology (KIT), and Forschungszentrum Jülich (FZJ). Here, the research is organised along the topics: a) stellarator research, b) tokamak physics – ITER and beyond, c) fusion technology for ITER, d) fusion technology after ITER, e) plasma-wall interaction, and f) plasma theory. This report presents results having been achieved by Forschungszentrum Jülich in the year 2010.

**Programme results – Highlight 2010**

The ITER-like Wall in JET – test of ITER’s burn chamber already today

Current research activities addressing the interaction of hot nuclear fusion plasmas with burn chamber walls aim at developing concepts for ITER and power plants beyond. Here, the chosen materials should show a long lifetime and a small fuel gas retention. At the same time, they must suffice the harsh environmental conditions of a burning fusion plasma, e.g. steady state neutron bombardment. As a first step, the materials being selected are of crucial importance. Based on experimental work at the ASDEX-Upgrade and TEXTOR tokamaks at Garching near Munich resp. Jülich as well as on experiments at special test facilities, Forschungszentrum
Jülich as a member of the Helmholtz Association of German research centres has taken over a leading international role in qualifying suited wall materials.

During the initial experiments on ITER it is planned to make use of a combination of Graphite and Tungsten in the heavily loaded divertor region while Beryllium is foreseen as the surface material for the rest of the burn chamber. Later, the divertor will consist of Tungsten only. Then, the properties of this new wall system will fully determine the operational regime of the fusion plasma.

It is therefore necessary to prepare and secure this starting phase of ITER operation by experiments already today. Hence, a new wall structure made of Beryllium and Tungsten is being integrated into today's leading European fusion experiment JET. The set-up will be very similar to ITER and shall be tested and explored under ITER relevant conditions. The new JET divertor consists of bulk Tungsten in its outer region. The system has been designed and built with major contributions from Forschungszentrum Jülich. It is based on a complex structure consisting of about 9,000 single Tungsten lamellae which are fixed by a chain of heat resistant Tungsten alloys and Nimonic. In parallel IPP Garching and other European fusion laboratories together have developed a special material made of bulk Graphite resp. CFC and a 20 to 30 μm thick layer of Tungsten deposited onto it. This will be placed at the remaining regions of the inner and outer JET divertor.

System integration into JET is nearly completed. Scientific experiments with the new "ITER-like Wall" will start mid 2011. Here, scientists from Jülich and Garching will play a major role. Forschungszentrum Jülich's leading expertise in Plasma-Wall Interaction research has once again been fostered and widened.
Nuclear Fusion Programme – Progress Report 2010

B.1. Plasma-Wall Interaction

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Introduction

The interaction of the fusion plasma with its surrounding walls (Plasma-Wall Interaction, PWI) is one of the major critical issues on the way to a fusion reactor and the major research topic in the Institute of Energy and Climate Research – Plasma Physics of Forschungszentrum Jülich. For plasma operation, plasma wall contact is a necessary condition to remove the heating power provided by external plasma heating and/or by the alpha particles and to remove the helium ash from the burning plasma. For the walls, the most important issues determined by PWI processes are the lifetime of wall components and the long term fuel (tritium) retention in the walls, which both critically affect the availability of a fusion power plant.

Generally, plasma-wall interaction is determined by the plasma properties in the vicinity of the walls, which depend both on the near wall plasma properties (temperature, density, magnetic configuration) and on material and surface properties (heat conductivity, erosion and re-deposition, hydrogen and helium uptake, thermal shock behaviour).

The actual activities are focussed on critical PWI questions defined for the operation of the ITER experimental fusion reactor which is under construction in France as an intermediate step to a fusion power plant. These questions are defined as a collaborative effort between our institute, the Task Force Plasma-Wall Interaction within the EU fusion programme (EU PWI Task Force, http://www.efda-taskforce-pwi.org) and the International Tokamak Physics Activity (ITPA). The main activities can be ordered and summarised as follows.

1. Wall erosion, material transport, re-deposition and fuel retention

These are the dominant processes determining the lifetime and long term fuel retention. For carbon PFCs, the work concentrated on erosion behaviour, in particular under ITER-like divertor conditions and on transport of carbon near plasma wetted surfaces and into gaps, which are foreseen to reduce the thermal stresses and possible cracking. This was accompanied by investigations of the fuel retention in bulk graphite, in particular under mixed species impact (He, Ar), and in thick carbon deposits. Part of the R&D on these topics included also the development of special tools to measure in situ the fuel retention in the wall components and the amount and composition of redeposited wall material, without breaking the vacuum and removal of wall tiles. These techniques are based on laser methods by which dedicated wall areas are heated to induce fuel desorption or to ablate material deposits together with spectroscopic
detection of released or ablated species. In parallel fuel removal techniques have been investigated further based on production and cleaning with special conditioning plasmas under the presence of magnetic fields and reactive gases such as oxygen and others. The production and analysis of these Ion Cyclotron Wall Cleaning discharges (ICWC) is followed up in close cooperation with the Tore Supra, Asdex Upgrade, JET and EAST tokamak.

In parallel, lab-scale experiments have been performed to assess the cleaning efficiency of Glow Discharge (GDC) and Electron Cyclotron Resonance heated (ECR) plasmas, in particular in the gaps of wall elements. Fig. 1 shows an ITER like castellated wall structure where the front and side wall areas have been coated ex-situ with a carbon layer to simulate carbon deposition after plasma operation. The layers have been removed in glow discharge and electron resonance heated plasmas in hydrogen and oxygen at 470 and 620 K. The cleaning efficiency in hydrogen plasma is low, while oxygen cleaning at 620 K can provide an effective method for cleaning. The cleaning shows a similar dependence on the depth of the gap as the deposition after plasma operation.

![Fig. 1:](image)

**Fig. 1:** Left: view of ITER like wall structures used for cleaning experiments in hydrogen and oxygen GDC and ECR plasmas. Right: comparison of deposit removal efficiency, normalized to the impinging plasma flux at the plasma-wetted top surface of the castellation, for cleaning discharges in oxygen and hydrogen at different temperatures.

### 2. Tungsten as Plasma Facing Component

Tungsten represents the most promising candidate as wall material for future fusion reactors and qualification of tungsten as PWI material is a main R&D issue in the worldwide fusion programme. Research at FZJ concentrates on the erosion and in particular on the high temperature behaviour of tungsten under melting conditions, such as melt layer stability, droplet formation and properties of resolidified tungsten. In addition, retention of hydrogen and helium in bulk tungsten and tungsten coatings is investigated.
The above figures show the ejection of tungsten fine sprays (left) and droplets (right) during melting of solid W samples mounted at test limiters in TEXTOR and a SEM view of the resolidified tungsten after a single melt event in TEXTOR. The experiments show that the melt motion under TEXTOR melt conditions is perpendicular to the magnetic field direction and dominated by the jxB force of the thermo-electron emission current with the toroidal magnetic field. The melting can cause W boiling leading to μm-sized fine W-spray into the plasma, occasionally accompanied by the ejection of larger droplets which can extinguish the plasma. Metallurgical observations show a strong material degradation, indicated by changes in grain size, dendritic growth and void formation. The behaviour of molten W under tokamak melting conditions is investigated in cooperation with Asdex Upgrade and the modelling of melt layer behaviour with the Karlsruhe Institute of Technology (KIT).

3. Material mixing

In ITER the use of three plasma wall materials (C, Be, W) will lead to a material mixing process, in particular in net material deposition areas, such as the inner divertor or the divertor dome region. This mixing can lead to new physical behaviour, such as possible alloy formation (e.g. be with W) resulting in a different material behaviour (e.g. reduction of the melting point) and influences in particular the possible retention of tritium by codeposition in these mixed layers. In the PWI group, special wall elements have been prepared consisting of W and C parts to induce the codeposition of eroded W with C. This intends to simulate ITER conditions and the retention of fuel (deuterium) in these mixed codeposits will be determined in the near future.

4. Transient heat loads

Plasma transients such as edge localised modes (ELMs), disruption and runaways represent a major challenge to the integrity of wall components in future fusion devices and require a high degree of avoidance and control. In our institute, scenarios by massive gas injection are under
investigation by which the effect of disruptions and runaways on the heat loads to the walls and the mechanical forces can be reduced.

In recent experiments, the effect of large type I ELMs on the target and the plasma radiation has been investigated in the JET tokamak using an improved bolometer diagnostic, demonstrating the production of large type I ELMS with plasma energy losses in the range 0.25-1.3 MJ.

Taking into account the radiation loss in the energy load calculation onto divertor target (red symbols in Fig. 3), the expected target load at an ITER-like collisionality of $v^{*\text{ped(neo)}} = 0.062$ would be 11 MJ (for $W_{\text{ped,ITER}} = 112$ MJ). For ITER conditions, this energy load requires a decrease in the ‘natural’ ELM size by a factor of $\sim 10$, assuming a wetted area in the inner/outer divertor of $\approx 1.2 \text{ m}^2 / 1.5 \text{ m}^2$.

**Fig. 3:** Normalized ELM energy loss and the target load for a range of type I ELMS in JET plotted against the edge collisionality.

Disruptions caused by vertical displacement events (VDEs) generate the largest radiative heat loads, with a maximum of the Radiation Peaking Factor (RFP) of about 3.5. The ‘ablation/melting parameter’, which determines the surface temperature rise caused by VDEs, can reach in ITER values up to 8.5 MW m$^{-2}$s$^{1/2}$ and will increase the beryllium temperature to values around $1/3$ of the melting point. It was demonstrated in dedicated experiments on Massive Gas Injection (MGI) for disruption mitigation that about 60% of the thermal energy and a significant part of the magnetic energy ($\approx 50\%$) was converted to radiation and spread uniformly over the walls. Nearly symmetric poloidal distributions of the radiation during precursor, thermal and current quenches have been observed (RPFs $\leq 1.5$). MGI-triggered disruptions are thus much less critical for ITER than VDE disruptions.
5. Plasma-wall interaction under cold (detached) plasma conditions and under helical 3D edge conditions

Detached plasma conditions (low temperatures < 5 eV and high density) are the strategy for ITER and future reactors to enable the compatibility of the divertor plasma with the walls on a long, steady state time scale. These conditions have been investigated by the FZJ group in the JET tokamak and will be also a major subject of the MAGNUM linear plasma facility. Also, 3D magnetic structures induced by external fields (resonant magnetic perturbations, RMP) are the present strategy to control ELMs in ITER. These structures also influence the plasma wall contact, mainly for the erosion and deposition behaviour. With the edge structure in TEXTOR induced by the dynamic ergodic divertor, these structures are simulated and its influence on PWI processes has been analysed. This research is also done in close cooperation with the DIII-D tokamak in San Diego/USA.

6. Behaviour of mirrors under erosion and deposition conditions

Mirrors positioned near first wall structures are inevitable to ensure an adequate diagnostic of plasma properties. Like other wall elements, these mirrors are subject to erosion and material deposition which degrades their performance. We investigate the physical conditions of these processes, their influence on the mirror performance and possible mitigation and cleaning techniques in close cooperation we the diagnostic working group (see also report on diagnostics).

Rhodium (Rh) and molybdenum (Mo) coated and single crystal (SC) molybdenum mirrors were tested under erosion conditions in edge plasmas of TEXTOR, showing that the single crystal mirrors preserved their optical properties, while Mo and Rhodium coated mirrors showed a drop of the reflectivity with a maximum value of 25% for the Rh-coated mirror.

Gas feeding in the vicinity of mirrors has been explored to mitigate the impurity deposition in a periscope-like system equipped with molybdenum mirrors and exposed in the edge plasmas of TEXTOR under deposition-dominated conditions. Both clean Mo and mirrors pre-coated with an amorphous carbon film (a-C:D) were used. With local helium gas feeding, the deposition on the surface of the first mirror was significantly suppressed, but still not enough to protect the mirror completely. After similar exposure with deuterium feeding, a full suppression of deposition on the first mirror and even complete removal of the pre-deposited a-C:D film were observed and the reflectivity of the mirror was restored, showing that local carbon deposition can be controlled to some extend by local hydrogen gas injection. The behaviour of mirrors is also investigated in cooperation with external partners in other fusion devices, e.g. Asdex Upgrade and DIII-D.

6. Qualification of atomic and molecular data

The study of plasma-wall interaction and the associated modelling requires a detailed diagnostic of the plasma boundary layer as well as good atomic and molecular databases for the inter-
interpretation. In the PWI group, a detailed study has been started to identify and absolutely quantify W0 and WI neutral and ionised emission lines of tungsten. This was done both with injection of WF$_6$ gas through limiters and by analysis of W sputter erosion in TEXTOR. This was accompanied by dedicated modelling of the tungsten spectroscopic emission with external partners (Efremov Institute, Russia).

7. Development of modelling tools

The PWI experimental investigations are accompanied by dedicated modelling of erosion, material transport and fuel retention. This is the basis to improve and validate the modelling tools and to extrapolate experimental results to conditions for ITER and other devices. One part of the work concentrated on modelling of experimental observations of local transport and redeposition of tungsten and carbon, with experimental data from WF$_6$ injection through limiters in TEXTOR, carbon deposition at remote areas in TEXTOR after remote CD$_4$ injection and material deposition in gaps of wall structures exposed under different conditions in TEXTOR. Another important topic is the possible effect of Be impurities and Be deposition on the chemical erosion of the carbon target in ITER. This is experimentally investigated in the PISCES linear plasma facility and modelled with the ERO code. Fig. 4 shows the reduction of the chemical erosion of graphite modelled with the coupled Tridyn surface and ERO erosion-deposition code under different Be seeding rates and assumptions on the surface Be-carbon reactions. The data show that the assumption of spontaneous Be-carbide formation resulting in an instantaneous insensitivity of the Be-bounded carbon against chemical erosion is needed to explain the experimental observations.

![Fig. 4: Modelled chemical erosion yield of graphite at different Be seeding rates in the linear plasma facility PISCES assuming a simple surface mixing (no BC) and with the assumption of an instantaneous Be-carbide formation.](image)

These data are finally used to perform and continuously improve the predictive modelling for the target lifetime and tritium retention in the divertor of ITER. The ERO code is applied in
cooperation with other partners to other experimental conditions in various tokamaks, such as Asdex Upgrade, JET, LHD and EAST.

8. Preparation of the ITER-like wall project in JET

In line with the preparation work for Plasma-Wall Interaction in ITER is the strong contribution of the Main Topic Group to the currently largest tokamak, JET, within the associated Task Forces E1 and E2 (“Exhaust”, please see the web link http://www.jet.efda.org/ and also http://users.jet.efda.org/tfwiki/index.php/Main_Page). At JET, a new plasma-facing wall is being installed. It shall be tested in the frame of the ITER-like Wall project from mid-2011. This will act as a test bed for the currently foreseen material options in ITER, especially beryllium and tungsten. The PWI group in FZJ has taken a large role in the technical preparation of this project by the leadership in the design of the bulk W outer divertor row, which has been completed in 2010 and is presently under installation.

Fig. 5: One of the solid tungsten divertor modules for JET, consisting of 4 stacks (poloidal direction) and individual W lamellas in toroidal direction which are electrically isolated against each other.

Beside this major effort, new diagnostics have been designed and procured, such as a new divertor observation endoscope, which are essential to enable an optimised diagnostic of the PWI processes at the walls (Be) and in the divertor (W) of JET. To monitor the material migration towards remote areas, which is expected to decrease largely with the elimination of graphite and the use of Be and W, new Quartz deposition monitors have been procured and installed at remote areas in the divertor.

The PWI work is organised in a topic-oriented manner with experimental work on a several facilities, depending on the actual conditions in relation to the topic of investigation. Among those, the TEXTOR tokamak is the major facility where PWI R&D is mainly done on dedicated wall components exposed in two PWI test facilities to the specific TEXTOR edge plasma conditions. A significant part of the scientific work was performed on the JET tokamak, embedded in the Task Force E (exhaust and edge physics) and other PWI elements have been investigated in DIII-D and Asdex Upgrade. The linear plasma device PSI-2 in our institute has
been assembled and commissioned and will contribute to the scientific PWI programme from 2011 on. The large linear high flux MAGNUM device (FOM, TEC), which will serve as a divertor simulator, is expected to contribute to the PWI programme from mid 2011 on, in the frame of the TEC collaboration.
Introduction

The activities of the Tokamak Physics Topical Group are strongly focused on areas where TEXTOR has unique characteristics and can make significant contributions for next generation fusion experiments, e.g. ITER, and improve physics understanding. The dynamic ergodic divertor (DED) is a unique tool to apply and study the effects of (rotating) resonant magnetic perturbations (RMP) with emphasis on the physics of field penetration, the applicability of RMPs to mitigate disruptions, and the effect of RMPs on plasma rotation and edge turbulence. Experiments in 2010 aimed at the characterisation of runaway electron beams which are generated during plasma disruptions and the exploration of massive gas injection to suppress or mitigate the effects of runaway electrons. A newly developed magnetic probe allowed the first direct measurements of screening currents on resonant flux surfaces which shield or weaken the amplitude of magnetic perturbations. The good accessibility and a set of complementary turbulence diagnostics (electric probes, spectrometry, reflectometry) allow detailed investigations of turbulent transport in the plasma edge. Further experiments aimed on the understanding of the mechanisms of intrinsic plasma rotation and the rotation braking due to neoclassical toroidal viscosity.

Runaway electron energy spectra during disruptions

In tokamak disruptions large electric fields occur due to a fast cooling of the plasma. These fields are capable of detaching plasma electrons from the thermal distribution and accelerating them to energies up to several tens of MeV. These runaway electrons present a threat for the machine as they can be dumped to the vessel wall or plasma facing components quite locally and hence cause severe damages. A scintillator probe was used to measure the energy distribution of runaway electrons in induced TEXTOR disruptions. The probe is shielded against the plasma by a graphite housing. Nine scintillating crystals, separated by stainless steel, detect electrons of different energies. Two thermocouples measure the temperature rise in the probe at the front and back side. The probe is inserted to the edge of the tokamak plasma and detects the absolute number, the energy spectrum, the temporal and spatial evolution, and the heat load of the runaway electron beam. For the measurement disruptions were induced by the injection of argon into the vessel what reliably produces a substantial number of runaway electrons. In the experiment the probe was positioned at various radial positions to check for a dependence of
the runaway spectrum on the radius. Furthermore measurements were done with inverted magnetic field and plasma current. The influence of resonant magnetic perturbation on the runaway losses was measured using the DED in 3/1 DC mode. The evaluation of this experiment is in progress.

**Influence of the DED on Spectral Runaway Distribution**

Resonant magnetic perturbations are a promising technique for runaway mitigation. At TEXTOR the DED has been operated in 6/2 mode which provides a strong ergodisation without exciting a tearing mode. The influence of the RMP on the runaway losses has been measured in low density discharges where a significant population of runaway electrons develops during the flat-top phase. The orbits of the runaways resemble the ones of the magnetic field lines. However, they are not identical because the runaways are displaced from the magnetic surfaces depending on their energy such that the orbits of the low MeV electrons are closer to the field line structure while the effect of ergodisation on high MeV runaways is strongly reduced. The scintillator probe is used to measure the runaways which leave the plasma. The probe is mounted on a drive mechanism which inserts it into the edge plasma shortly before the DED is switched on. The time evolution of probe signals for different runaway energy ranges is shown in figure 1. The dashed lines outline the period of the probe being inserted into the plasma and the dotted lines outline the phase in which the DED operated at constant perturbation current of 6 kA with a ramp up and down before and after, respectively. With the onset of the magnetic perturbation the probe signal increases; the losses of runaway electrons grow with the ergodisation level. Also details of the spectrum of the lost runaways change from low to high ergodisation level.

![Fig. 1](image_url): Measured runaway electron losses at various energies during a runaway discharge where a magnetic perturbation is applied.
Edge magnetic topology measurement using a fast movable magnetic probe

Plasma response to resonant magnetic perturbation fields (RMPs) is important for understanding the physical mechanism of instability control using RMP fields in the next generation of fusion devices, i.e. ITER. To date, many attempts to explain ELM suppression/control using RMP fields have focused on the idea that the edge thermal and particle losses are enhanced due to formation of an ergodic zone with RMP fields. The ergodic boundary would reduce the edge pressure gradients and thus stabilise the peeling-ballooning modes thought to underlie ELM formation. However, either bulk plasma or diamagnetic rotation can screen the RMP fields from the resonant magnetic flux surface. Many calculations of the Chirikov parameter or overlapping of resonant magnetic islands employ a vacuum assumption which neglects the plasma response due to rotational screening and modification of the underlying equilibrium.

**Fig. 2:** Photograph showing the magnetic probe (graphite shielding removed). Each of the three sets of coils measures the magnetic field in three orthogonal directions.

On TEXTOR, investigation of plasma response to RMP fields has been carried out under different dynamic ergodic divertor configurations. The fast movable magnetic probe (FMMP) has been installed at the outer equatorial plane (low-field side). Three sets of coils are radially mounted at the probe head with a distance of 0.5 cm as seen in figure 2. Each set of coils includes three coils measuring the local poloidal, radial and toroidal magnetic fluxes, respectively. By integrating the measured local magnetic fluxes, the local magnetic field in three orthogonal directions can be obtained. During the application of RMP fields the perturbed magnetic field can be measured. Preliminary results show that the perturbed plasma edge magnetic topology is different from the case simulated with a vacuum assumption. Plasma response to RMP depends strongly on both the location of the resonant rational flux surface and the frequency difference between the drift of the rational surface in the plasma and the external perturbation. Figure 3 shows an example of modelling a synthetic signal measured with the
FMMP in presence of a 5 kHz co-current direction rotating dynamic magnetic perturbation. It is the first time that a narrow helical current sheet (also referred to as plasma screening current) is observed at the resonant magnetic flux surface before the field penetration occurs.

Fig. 3: Modelling of a synthetic signal of the fast magnetic probe in presence of a dynamic magnetic perturbation.

Neoclassical toroidal viscosity (NTV) braking

The NTV torque induced by a non-axisymmetric magnetic perturbation (NAMP) in the collisionless regimes in tokamaks is obtained by numerically solving the bounce-averaged drift kinetic equation. In different asymptotic limits of the collisionless regimes, the numerical solutions are in good agreement with the analytic results. The analytic results show deviations from the numerical modelling in the transient regimes. The numerical method can be applied for modelling the NTV torque in different collisionality regimes and the transition regimes in tokamak plasmas without using additional approximations. The effect of resonant particles makes the NTV torque more important at lower collisionality and lower rotation, which are the ITER relevant conditions.

There is no obvious braking effect with $m/n = 6/2$ magnetic perturbation field applied by the DED for both resonant and non-resonant helicities before the field penetrates into the plasma. The calculated NTV torque on TEXTOR is also very small, because of the fast decay of the perturbation field inside the plasma.
Influence of resonant magnetic perturbation and plasma density on the long-range correlation and zonal flows

Long-range correlations (LRCs) of plasma turbulence and associated zonal flows have been routinely observed at TEXTOR tokamak using two sets of Langmuir probe systems. However, the impact of magnetic topology and plasma parameters (e.g. collisionality) on the LRCs is not yet clear. To this end, we applied the TEXTOR-DED and varied the plasma density to investigate the influence of (i) resonant magnetic perturbation and (ii) plasma density (collisionality) on the LRCs and related zonal flows. In TEXTOR, the dynamic ergodic divertor can ergodise the edge magnetic field lines and thus create a stochastic magnetic topology via the RMP at the plasma boundary. With increasing DED current, the ergodisation is expected to be stronger. It has been found that the LRC detected by two distant probes gradually reduces with increasing DED current (figure 4). The results reveal a damping effect of the RMP on the LRC and related zonal flows. A possible mechanism for such effects is due to the modification of parallel dynamics of zonal flows by the RMP. The results may provide an explanation for the absence of the LRC in the RFX-mode reversed field pinch experiments, where the edge island chains increase the magnetic stochasticity and thus limit the development of zonal flow structures. To explore the impact of the plasma density (or collisionality) on the LRCs, we gradually increased plasma density in the proximity of the density-limit at TEXTOR. In low density cases the LRCs are quite large and change very slightly with increasing density. However at higher densities when approaching the density-limit, the LRCs decrease rapidly with increasing density. The increase of plasma density usually induces a reduction of edge temperature and consequently a change in collisionality. It is interesting to find that the reduction of the LRC due to increasing density is always accompanied by a reduction of mean radial electric field $E_r$. Therefore, the results suggest the possible role of collisionality and the impact of mean $E_r \times B$ flow shear on the long-range correlation as well as on zonal flows.

Fig. 4: Suppression of long-range correlations by the RMP. Maximum values of the LRC measured by two distant probes under three different DED currents in an m/n=6/2 configuration.
Physics of edge localised modes (ELM) in limiter H-mode plasmas

Recently, multiple resonances in the ELM frequency as a function of the edge safety factor have been observed for the first time with an applied low \( n \) resonant magnetic perturbation field on JET. This experimental result suggests that there are two effects of the RMP on the ELM frequency: a global effect and the multi-resonance effect. The RMP global effect, which has no safety factor dependence, results in a relatively weak increase of the ELM repetition frequency. In contrast to the global effect, the RMP multi-resonance effect depends strongly on the edge safety factor and causes a stronger increase of ELM frequency. These two effects are most likely due to different physics mechanisms. A model which assumes that the ELM width is determined by a localised relaxation triggered by an unstable ideal external peeling mode can qualitatively predict the observed resonances when low \( n \) fields are applied.

In limiter H-mode plasmas on TEXTOR the ELM frequency is about few hundred Hz, which is by one magnitude higher than the Type-I ELM frequency, and similar to the Type-III ELM frequency observed normally in X-point divertor devices. Recently, experimental investigation of the edge safety factor dependence of the ELM frequency has been performed. Preliminary results show that the ELM frequency depends strongly on the edge safety factor and is very reproducible. By a small change in safety factor \( q \) from 3.4 to 3.6, the ELM frequency was reduced by more than a factor of 2 from \(~1\) kHz to \(~400\) Hz. A minimum in frequency appears at \( q = 3.6 \) when \( q \) is varied in the range from 3.4 to 3.9.

Using a Taylor relaxation theory, initialised by an external peeling mode, the widths and frequencies of ELMs can be modelled. The dependence of the ELM frequency on the edge safety factor has been investigated and a development of multiple resonances at lower values of the normalised edge current density has been found. Taking into account small variations of the plasma current allows to explain the experimentally observed range of ELM frequencies with this model.

Edge turbulence during the ELM cycle

Edge Localised Mode (ELM) manifest itself in cyclic significant variation of plasma density and temperature profiles at the plasma edge accompanied by a rapid ejection of energy and particles onto plasma-facing components. Periodical modifications of plasma temperature and density gradients leads to modifications of stability properties in this region. The latter one is a possible reason of the observed variation of the plasma fluctuation spectrum measured with reflectometry during the ELM cycle. In between two consequent ELM crashes the turbulence level (figure 5) is often observed to decrease to very low amplitude (usually observed in plasmas without auxiliary heating). We refer to this period as 'Silent Stage' when phase fluctuations in reflected signal are so low that the phase trend can followed and reflection position movement can be easily reconstructed. Reduced amplitude of plasma perturbations during a 'Silent Stage' leads to more coherent reflection of probing waves and as a consequence to a larger amplitude.
Plasma rotation due to resonant magnetic perturbations

Experiments have been carried out in order to investigate the mechanisms responsible for driving plasma rotation by external resonant magnetic perturbation field. During the application of a static (dc) RMP field with $m/n = 6/2$, the rotation change is observed to be in the ion diamagnetic drift (IDD) direction. This indicates that the RMP field exerts a torque in IDD direction on plasma. When low frequency ac fields, rotating in electron diamagnetic drift (EDD) or IDD direction, are applied the rotation change is also in IDD direction. This torque in IDD direction is considered to be the stochastic torque generated in the edge ergodic region where a positive radial electric field builds up due to the larger transport of electrons than ions.

When an ac field with $m/n = 3/1$, rotating in the EDD direction, is applied, the rotation change is observed in the EDD direction. This torque in EDD direction cannot be explained by the edge stochastic torque which is always in the IDD direction. In another experiment, when the plasma current is reversed so that the external field is not aligned with the pitch angle of the magnetic field, the plasma rotation is not influenced. This proves that the rotation change is produced by a resonant interaction between plasma part external field. A possible mechanism for the observed EDD torque is that it is induced by the shielding current on the rational surface when the RMP field is applied.

Intrinsic Rotation

The observation of an additional torque in plasmas after application ion cyclotron resonance heating (ICRH) is considered to be very important for future generation of tokamaks like ITER and presently under investigation on many fusion devices. At TEXTOR, such a study is unique because of the availability of (i) advanced diagnostics for plasma flow measurements such as correlation reflectometry and charge exchange recombination spectroscopy, and (ii) auxiliary
heating systems such as ICRH and neutral beam injection (NBI) in co and counter current directions. At TEXTOR, ICRH applied in the plasma centre leads to an acceleration of plasma rotation both in electron diamagnetic drift direction and in toroidal counter-current direction. It was found that (i) the effect increases with ICRH power, (ii) higher NBI power injected in co-current direction leads to a larger effect, and (iii) the radial electric field in the outermost plasma region is found to decrease gradually with ICRH power in accordance with neoclassical estimations.

**Collective Thomson scattering measurements of fast ion redistribution due to sawteeth**

The study of the fast-ion dynamics during sawtooth oscillations was continued on TEXTOR in 2010 using the CTS diagnostic in neutral beam injection heated plasmas. The study indicates that the redistribution of the fast-ion population in the plasma centre at the time of a sawtooth collapse is pitch angle dependent. Outside the sawtooth inversion radius the measured fast-ion dynamics show no correlation with the sawtooth oscillation. This effect is illustrated in figure 6 where the fast-ion dynamics is shown for different projection angles and radial positions.

**Fig. 6:** Projected fast ion density as a function of time for 5 different discharges with similar sawteeth. A sawtooth crash in the electron temperature is illustrated with a vertical dotted line. Clear sawtooth oscillations are only seen in discharge 106737 where the measuring volume is located in the plasma centre.
Measurements during the sawtooth ramp-up, for specific sawtooth oscillations, indicate that the density of fast ions with energies close to the full injection energy is reduced before the time of the sawtooth collapse. This effect can partly be explained by changes in the classical slowing down time during the sawtooth oscillation.

**Fast ion dynamics and comparison with simulations**

Collective Thomson scattering (CTS) allows measurements of the fast ion velocity distribution function in the plasma with spatial and temporal resolution. The goal of the CTS experiments was to obtain the fast ion distribution at different positions and angles to the magnetic field, and compare the measurements with numerical simulations. The experiments were done in low density deuterium plasmas. The source of fast ions was one neutral beam injector which injected hydrogen atoms at an energy of 50 keV. The measurements were made at the plasma centre around \( R = 1.8 \) m and off-axis at \( R = 2.0 \) m, each for two different angles with respect to the magnetic field lines.

Simulations were made by the guiding centre Monte Carlo code ASCOT and the drift-kinetic Monte Carlo code VENUS. The two simulation codes disagree in the plasma centre and agree well off-axis. This may be due to the different neutral beam ionization models.

The comparison of the measured fast ion distribution and the simulated velocity distribution functions show good qualitative agreement. In TEXTOR where the scattering volume is located in the plasma centre one finds an excellent agreement between the measured and simulated distribution functions. Within the range of investigated plasma parameters the fast ion velocity distribution function in the plasma centre can be described solely by collisions. However, the off-axis measurements show a large discrepancy between the measured and the simulated fast ion distributions. The experimental results have significantly higher values than the simulated ones. The measurement therefore suggests a larger population of fast ions at the off-axis location compared to the simulations. Part of this discrepancy could be explained by the uncertainty of the radial position of the scattering volume. The discrepancy cannot, however, be entirely explained by such misalignment. The fact the fast ion profile seems to be more peaked in the simulations compared to the measurements may also in part be due to additional outward transport of fast ions not accounted for in the codes.

**Excitation and suppression of Alfvén eigenmodes (AEs) by RMP**

Alfvén eigenmodes (AEs) were excited by applying a small rf current (< 4 A) on the DED coils. The effects of the edge magnetic perturbation on AEs have been studied by superimposing the dc current on the DED coils and preliminary results on suppression of AEs by DED fields were demonstrated. In addition, NBI fast particle induced TAEs were suppressed by application of \( m/n = 3/1 \) DED fields. The externally excited AEs for various plasma parameters (plasma current, toroidal field, plasma density) were characterised and a reasonable agreement with the theoretical prediction (figure 7) has been found.
Fig. 7: Comparison between measured and calculated AE frequencies.
Introduction

Apart from understanding physics aspects of fusion relevant plasma, many challenges in terms of fusion technology have to be met on the way towards commercial power generation from nuclear fusion. The scope of the Topic Group Technology is to deal with engineering science and technology aspects of projects in fusion. This covers all phases such as concept, design, manufacturing, testing, and commissioning of new experimental devices, as well as operation, service, and upgrade of existing experimental devices. The selection and implementation of common tools and standards required for technology projects also form a part of the Main Topic. Currently, the main focus is on contributions to fusion experiments such as ITER, Wendelstein 7-X, JET, TEXTOR, and to material test facilities such as PSI-2, MAGNUM, and MARION. The final goal of all fusion technology projects is to provide new experimental capabilities and horizons. The plasma physics background of the relevant projects is therefore described in the corresponding Main Topics.

ITER Core Charge Exchange Recombination Spectroscopy

The core Charge Exchange Recombination Spectroscopy (CXRS) is a diagnostic system that allows determining several plasma parameters by evaluating spectral lines in the light generated by charge exchange recombination processes due to interaction of a neutral particle beam with the plasma. The corresponding diagnostic neutral beam injector (DNBI) is under responsibility of India. The edge CXRS system provided by Russia is located in the equatorial port no. 3, while the upper port no. 3 accommodates the European core CXRS which is the subject of this section. An overview on the ITER CXRS diagnostic system is shown in figure 1.

Concept studies of the system are developed by a cluster of research organisations which is jointly led by Forschungszentrum Jülich and ITER-NL with participation of the EURATOM associations CCFE and HAS. The Jülich Main Topic Technology is developing the port plug conceptual design while ITER-NL provides substantial contributions in the fields of project engineering, optical design and neutron analysis. The partners of the cluster have prepared a consortium agreement with the aim of coordinated bidding for F4E grants and coordination of development activities for ITER CXRS.
In an earlier EFDA task a detailed project plan for the full development and implementation of the core CXRS diagnostic for ITER has been developed based on a reference concept of the port plug lay-out. Supported by additional national project funding with the aim to prepare for ITER contracts, the major partners of the cluster ITER-NL and FZJ are working on development and prototyping of critical components of ITER diagnostics. In the field of mechanical engineering the main focus is on the conceptual design of the port plug and its components. The port plug has general functions such as maintaining the vacuum integrity and the tritium confinement of the ITER vacuum vessel, and providing neutron shielding to the toroidal field coils as well as to exterior components. The CXRS specific function is to accommodate the optical labyrinth. The first mirror from plasma side is subject to substantial particle and radiation load. In order to maintain its optical performance special features for protection, cleaning, and replacement have to be foreseen.

In 2010 the reference concept has been developed to the level of advanced conceptual design. The compatibility with the new generic port plug shell has been analysed. Due to the fact that the interface to ITER machine and some of the physics parameters are not fixed yet, the activities are now focussed to the development of generic port plug components rather than a detailed port plug. In parallel an alternative port plug lay-out has been drafted in order to maximize the life time of the first mirror which is crucial for the diagnostic performance.

**Fig. 1:** Overview of the core Charge Exchange Recombination Spectroscopy.
The reference concept of the port plug is shown in figure 2. A cassette is inserted into the outer shell from the back side. It supports the retractable tube (RT) and the secondary mirrors forming an optical path. The cassette is attached to the outer shell by four keys (with sliding contacts at the nose) and is fixed to the outer shell rear flange using the vacuum weld. The RT carries the first mirror (M1) and the shutter, and allows multiple M1 replacement. The RT can be mounted and dismounted from the back side of the cassette using a special remote handling tool. The shutter, consisting of flexible arms bent by a pneumatic cylinder, protects M1 and has a diffuser on the rear side in order to reflect the light of the lamp to the radiometer and to the mirror labyrinth for calibration purposes.

![Fig. 2: Reference concept of the port plug.](image)

The compatibility of the CXRS reference port plug concept with the new generic port plug shell resulted in the general feasibility. However a few major modifications are required. Since the new shell is reinforced by an intermediate flange, the cassette has to be inserted from the front and adapted to this procedure. For reduction of transient electromagnetic loads the shielding cassette has to be split vertically. For stability reasons all optical components have to be attached to a single diagnostic shield module (DSM).

The results of detailed neutron, electromagnetic, thermal, hydraulic, and structural analysis have been used in order to optimize port plug components such as the shutter, the M1, and the RT (figure 4) with respect to performance and load carrying capability. The operating principle of the frictionless shutter actuator has been patented.

In particular, manufacturing methods have been studied for integration of the cooling lines into the retractable tube. The compliance of two options, rectangular cooling lines brazed into groves and cooling lines formed by explosion welding from metal sheets, have been verified for compliance with thermal requirements.
**Fig. 3:** Compatibility of the CXRS reference port plug concept with the new generic port plug shell including split diagnostic shield module (DSM) and diagnostic first wall DFW.

**Fig. 4:** Optimized port plug components. Shutter with actuator (top left), first mirror (top right), nose (bottom left), and flange (bottom right) of retractable tube.

For manufacturing reasons, the optimized shielding cassette (figure 5) consists of five sections, each of them having an individual water supply and a steel to water ratio optimized for neutron shielding.
Due to the predominant effect of the impurity deposition on M1, which has been shown by a performance model of the overall diagnostic system, several mitigation and cleaning methods have been studied. Considerable improvement could be achieved by reducing the solid angle of M1 towards the plasma. An alternative concept with the blanket aperture at the first wall rather than at M1 was found. In an iterative approach of a parametric CAD model of the port plug lay-out and an optics model a corresponding port plug concept has been developed. New feature apart from a reduced solid angle are a cleaning system for M1 in RT while M1 itself and shutter are no longer in RT, and a vacuum boundary moved to the intermediate flange in order to allow for a small ITER standard vacuum window.

**Fig. 5:** Optimized shielding cassette with five sections.

**Fig. 6:** Comparison of results from three models of the total torque on the port plug in case of a disruption.
In support of the port plug concept development numerous simulations have been performed. Three independent global electromagnetic models of a 20 degrees sector of ITER have been developed. In a benchmarking process models have been corrected resulting in a very good agreement of the results (figure 6) providing necessary input for disruption load simulations of both the port plug and its individual components.

Other simulations have been performed by contractors under supervision of FZJ. For example, the hydraulic and thermal simulation of the shielding cassette resulted in a redesign of the cooling line structure (figure 7).

![Cooling system of the front section of the cassette redesigned according to simulation results.](image)

**Fig. 7:** Cooling system of the front section of the cassette redesigned according to simulation results.

The overall target of the project is to develop concepts of the ITER CXRS port plug components proven by full analysis and prototype testing in order to be prepared for Fusion 4 Energy calls for grant proposals.

**Superconducting busbar system for Wendelstein 7-X**

The stellarator is the most promising alternative to the tokamak because of its inherent stationary plasma operation. The prospect of stationary operation opens new possibilities to investigate reactor-relevant physics issues. However, it also requires additional solutions for the accompanying technical problems which are related to the superconducting coils, the durability and cooling of wall elements as well as the control and data analysis of diagnostics. FZJ participates in the design and construction of the stellarator Wendelstein 7-X – which is presently under construction in Greifswald/Germany – by taking over essential work packages. During later operations, FZJ will also participate in the scientific analysis of the experimental results. FZJ contributes to both the machine itself as well as to diagnostic systems.
The superconducting busbar system project being by far the largest contribution to the machine construction has been successfully finalized by the assembly in 2010. In parallel to the assembly, support on structural simulations for verification of on-site modifications has been provided. After final assembly a full structural analysis of all bus modules including supports has to be done for verification. A sophisticated finite element model (figure 8) using beam and shell elements has been developed in order to minimize computing time thus allowing for multiple iterations of design and analysis.

Fig. 8: Finite element model of the Wendelstein 7-X busbar system.

Several contributions in the field of diagnostic development are already running while others are under way.

**Bulk tungsten divertor for JET**

The European tokamak experiment JET is to be equipped with a first wall which represents the material mix foreseen for ITER. FZJ contributes to this by the development of a bulk tungsten divertor module which is located in the most loaded area in the divertor. The full “LB-SRP” row – for ‘Load-Bearing Septum Replacement Plate’ – consists of 48 modules to cover the belt circumference. In a phase of intense R&D on a structure which can cope with expected electromagnetic loads during disruptions and with appropriate thermal loads, the final design has been proven by analysis and prototype testing. Series manufacturing has been almost finished.
in 2010. First modules have been equipped with diagnostic components and afterwards used for remote handling assembly tests in order to avoid any clashes during final assembly which is foreseen to be finished in spring 2011.

![Adaptor](image)

**Fig. 9:** Three JET bulk tungsten divertor modules in a row for remote handling assembly testing, the middle one still to be mounted showing the adaptor plate below.

In parallel to manufacturing and assembly, several new thermal simulations specifically for the cool down phase under various pressure values and further heat load exposures in the MARION facility have been performed in 2010.

**Linear plasma generators PSI-2 and JULE-PSI**

In 2010 the linear plasma generator PSI-2 (figure 10) has been constructed at FZJ with the first argon plasma achieved.

![Linear plasma generators](image)

**Fig. 10:** The linear plasma generator PSI-2 (left) and the first plasma (right).
The design of a target chamber with manipulator dedicated to the analysis of exposed samples has been started.

Another plasma generator (JULE-PSI) is to be constructed in the hot materials lab (HML) in order to allow exposure and analysis of activated and contaminated samples. The concept of this machine has been drafted to define the necessary infrastructure which is to be installed during refurbishment of the HML.

**TEXTOR**

The Jülich tokamak TEXTOR has been operated for the scheduled period mainly for experiments in the field of plasma wall interaction with an overall availability of about 90%. The performance has been improved by implementing a vertical drift feed-back control using evaluation of the light emitted from limiters. In addition several control systems have been upgraded. In the regular summer shutdown new (e.g. gas puff imaging) and improved versions (e.g. dispersion interferometer) of diagnostic systems have been implemented in order to extend the experimental capabilities.
Introduction

The development of plasma diagnostics and heating systems is a major topic of research performed by the TEC partners (FZJ, FOM and ERM/KMS), since these systems are essential to study the plasma properties experimentally (diagnostics) or to modify the plasma properties locally or globally (heating). New experimental discoveries in fusion research are often facilitated or even enabled by the successful development of new diagnostic or heating schemes.

The development of plasma diagnostic and heating systems is in itself an interesting and challenging field of work, with interfaces to plasma physics, applied physics and technology, where the underlying principles of diagnostic and heating systems are often related to basic physics processes and in particular to fundamentals of plasma physics.

The scope of the TEC Main Topic Group “Diagnostic and Heating” is to cover the scientific aspects of the diagnostic and heating development work described above, comprising the underlying physics principles, the technical realisation, assessment of performance and calibration issues. Several of the diagnostic or heating systems described below have already started to produce valuable physics results which are presented in the “PWI” and “Tokamak Physics” chapters of this annual report. Engineering aspects are described in the “Technology” chapter of this annual report. The annual report of this group is structured according to the following scheme:

- ITER diagnostics
- Wendelstein 7-X diagnostics
- JET diagnostics
- TEXTOR diagnostics
- Plasma heating

ITER diagnostics

The main ITER diagnostic project pursued by FZJ is the development of the core Charge Exchange (cCXRS) Diagnostic system, together with consortium partners from the Netherlands (ITER-NL), United Kingdom (CCFE) and Hungary (HAS). The cCXRS diagnostic is designed to allow determining local values of several important plasma parameters by evaluating spec-
tral lines in the light generated by charge exchange recombination processes due to interaction of a neutral particle beam with the plasma: First, the Helium density in the plasma, which is the product of the fusion process and hence an indicator of the success of the fusion process. Second, the ion temperature and velocity, together with magnetic field components, and the fuel ion ratio (D/T ratio) can be deduced from the analysis of the cCXRS spectra. Finally, the mean ion charge (Zeff) can be derived from the background level within the spectra.

Core CXRS diagnostic

The main work on cCXRS performed by FZJ within 2010 comprises the following sub-topics: First the development of concepts and components for the cCXRS upper port plug. This work is described within the technology chapter of this report. Second, a prototype spectrometer has been developed and tested successfully together with the partner ITER-NL. Finally, atomic data for an improved accuracy of spectra analysis from charge exchange with Argon ions and beam emission radiation have been calculated. In close relation to the development of optical diagnostic schemes, the lifetime optimisation of first mirrors under ITER conditions and mirror surface cleaning by plasma sputtering have been studied. An overview of the cCXRS diagnostic system was presented at the SOFT conference 2010 [W. Biel et al., Fusion Engineering and Design 2011, in press].

The new cCXRS prototype spectrometer was developed with the aim to allow for simultaneous measurement of spectra within three different wavelength ranges and at efficiencies (product of etendue and transmission) which are altogether matching the requirements for ITER. The overview on the spectrometer setup is shown in fig 1.

![Fig. 1: Setup of the ITER cCXRS prototype spectrometer [M. Durkut et al, SOFT conference 2010]](image-url)
A first test of a pre-version of the spectrometer was performed on TEXTOR in summer 2010 in order to obtain early results and first experience on the spectrometer operation and development status. From these first test spectra from TEXTOR it was found that both efficiency and spectral resolution for the Helium channel already met the target, while the spectral channels for Hydrogen (Hα) and Carbon needed further optimisation with respect to spectral resolution (alignment) and suppression of higher order (ghost) lines. This work on spectrometer completion and optimisation was successfully performed by the partner ITER-NL later in 2010 and a new test was being prepared for 2011.

**Development in atomic data for the ITER CXRS Project**

The quality of atomic data plays an important role in the measurements of impurity densities using the charge-exchange spectroscopy (CXRS) diagnostic. The density of impurities depends on two sets of atomic data, namely, the effective rate coefficients of charge-exchange recombination and rate coefficients of beam-emission spectra of Hα line.

The accurately calculated effective rate coefficients based on the new charge-exchange cross sections demonstrate a significant deviation from available ADAS data. The reason to such discrepancy is attributed to the influence of the cascades from the highly excited H-like states considered in the present model. On the other hand the collisional-radiative model of the hydrogen beam in parabolic states [O. Marchuk et al, Nucl. Instrum. and Methods 2010] was benchmarked against accurate measurements from the tokamak JET for magnetic components of the Stark multiplet. In Fig. 2 the results of the comparison are presented. An excellent agreement is found between the model and the experimental data for the ratio of σ- and π-components. The present comparison clearly demonstrates the advantage of the new collisional radiative model with respect to the statistical results and therefore it can provide reliable predictions for ITER plasma conditions.

**Fig. 2:** Comparison between measurements and results of the collisional radiative model: grey points-experimental data; solid lines – present calculations; dashed line-statistical expectation
First mirror investigations

Thermo-oxidation is one of the promising techniques for tritium removal from carbon-based deposits in ITER. However, there is a strong concern that thermo-oxidation may impact sensitive diagnostic components like mirrors. To investigate possible collateral effects of thermo-oxidative wall conditioning, diagnostic mirrors were exposed in the DIII-D tokamak during an O-bake experiment. Molybdenum (Mo) and copper (Cu) mirrors were exposed for more than 2 hours in an oxygen-helium atmosphere (20% O₂ + 80% He) at temperatures 160 °C and 350 °C and in-vessel pressure of 13 mbar. Strong surface oxidation and corresponding drop of the reflectivity in ultraviolet (UV) and visible (VIS) wavelength ranges were observed. A decrease of the specular reflectivity of Mo mirror was as high as 50%, while in case of Cu mirror it even has reached 90%. Our studies explicitly demonstrate a strong adverse impact of the proposed thermo-oxidative wall conditioning technique on diagnostic mirrors. Measures on the mirror surface recovery must be applied to the diagnostic mirrors exposed to thermo-oxidative conditioning to continue the feasibility assessment of O-bake in ITER. In particular, an effective in-situ cleaning system must be developed and implemented in the design of current mirror-based ITER diagnostics.

Tritium retention diagnostic

The second ITER diagnostic project comprises the development of methods for the measurement of the Tritium content stored within the first wall, material deposits and dust within the ITER vessel. Several laser aided methods are under investigation: First, the laser induced desorption spectroscopy (LIDS) is based on the local heating of a wall element by high power laser pulse (E = 60 J, t = 10 ms). Due to the temperature increase, the stored fuel atoms are driven out of the bulk wall material, entering into the tokamak plasma, which can then be detected by spectroscopic means. Within 2010, the detection limits for LIDS measurements could be significantly improved down to < 10¹⁷ D atoms per cm² by introducing a gated camera into the detection system. The second method is the laser induced ablation spectroscopy (LIAS), which uses a shorter laser pulse (~ 7 ns) with high pulse energy (1 – 2 J), thereby ablating a fraction of material from the bulk wall element. The atoms entering the main tokamak plasma are again detected and analysed using a spectrometer. Within 2010, first test measurements on LIAS performed in the visible and near UV spectral range. Additionally, an ITER-like coaxial observation system has been designed and constructed for a test of this detection principle on TEXTOR. Finally, the laser induced breakdown-spectroscopy (LIBS) uses an even higher laser power density which is sufficient to ignite a plasma with the ablated particles from the wall element, where the particle detection is then based on the spectroscopic analysis of the laser induced plasma. First laboratory experiments with LIBS were conducted in 2010 and a lower detection limit of 10¹⁸ C atoms was found [A. Huber et al, SOFT conference 2010].
Diagnostics for Wendelstein 7-X

Dispersion interferometer

In 2010 the dispersion interferometer (DI) project at TEXTOR made a large step forward. In early 2010 one DI module was operated for the first time in its final configuration. In summer 2010 four modules were operated in parallel as a multi-channel diagnostic. During about 150 shots in total so far, the system could prove its functionality, and a successful restart without problems was done after the summer break of TEXTOR.

Several measurements were performed piggy-back during regular TEXTOR operation, the expected accuracy – $3 \times 10^{17}$ m$^{-2}$ for the signal-to-noise ratio and 4 µs time resolution – could be achieved. One of the most impressive results is shown in figure 3: The evolution of the electron density for a disruption. During this event, a beam of runaway electrons was formed and could be detected with all four channels of the DI. It can be seen that the beam moves from the high field side of the plasma vessel to the low field side.

In addition to the completion of the four channel set-up, new in-vessel mirrors were installed inside TEXTOR during the maintenance phase in summer 2010. With these mirrors it will be possible to guide beam lines from the central part of the plasma to the edge regions, which is necessary to measure the density distribution and to control the plasma position.

After a successful test of the four channel set-up, the contract with Budker institute Novosibirsk could be finalized, the cooperation will be continued to improve and extend the DI system. Because of the successful test of the diagnostic, the Max Planck Institute in Greifswald (IPP) is highly interested to build a DI for the use at the Wendelstein 7-X experiment; for this, the cooperation between IPP and IEK-4 will be intensified.

![Figure 3](image-url)  
**Fig. 3:** Disruption detected with the dispersion interferometer at TEXTOR (shot #113399) with four channels at different radial positions. After the disruption at 4010 ms, a runaway beam is formed and detected by the DI at different times, i.e. the beam was moving from high field side to the low field side. Finally, the beam breaks down by inducing a final plasma pulse at 4040 ms.
Diagnostics for JET

Plasma edge spectroscopy

In 2010 the ongoing work on the upgrading of plasma edge and divertor spectroscopy has been almost completed. The main work packages comprised the design and procurement of a new KL11 divertor endoscope for the measurement of tungsten erosion in the new full W divertor of JET, the upgrading of the KL1 main chamber endoscope, the KS8 system as a main chamber impurity monitor and the fast KS3 divertor flux monitor. The KL11 system is designed for a wavelength range from 350 nm until 1700 nm, with an optimisation to detect the W I emission line at 400.8 nm simultaneously with Be II emission at 527 nm, C III emission at 465 nm and Dα emission at 656 nm in order to provide effective erosion yields from the spectroscopic signals.

TEXTOR diagnostics

Fast magnetic probe

A new fast magnetic probe has been designed and constructed to measure magnetic fluctuations in the edge plasma with high spatial and temporal resolution. The system consists of an array of 3 times 3 small magnetic pick-up coils with a size of 5 mm x 25 mm x 25 mm and orientation in all three spatial axes to measure radial profiles of the time derivative of the local edge magnetic field during experiments on TEXTOR with the dynamic ergodic divertor. The system can be mounted onto the fast reciprocating probe system which allows a fast (0.8 m/s) forward and backward movement of the system for a short time into the edge plasma of TEXTOR without excessive heat loads onto the probe head, thereby avoiding melting or destruction of the probe head during operation. First successful experiments were conducted and results are reported in the tokamak physics chapter of this report.

Gas puff imaging system

A new gas puff imaging system has been designed and installed on TEXTOR by our Belgian TEC partner ERM/KMS. The system consists of a fast camera which imaged the plasma edge region via a telescope. Injecting neutral gas towards the plasma edge region in front of the observation system, the spatial variations of light emission can be monitored with high temporal resolution and information on fluctuations of plasma temperature and density can be derived from the measured images.

He beam diagnostic

Within the frame of a collaboration with ORNL (USA), an improved He beam diagnostic system has been designed and constructed in front of the ICRH antennae at TEXTOR with the aim to facilitate experimental studies on the interrelation between local plasma densities and
ICRH power coupling. The system consists of additional He gas puff capabilities in the vicinity of the antennae and a new 12-channel filter-scope observation system which covers 4 different wavelengths, allowing for simultaneous determination of electron density, electron temperature and neutral particle densities in front of the ICRH antenna. The system was completed by end 2010 and a test for early 2011 was under preparation.

**Plasma heating**

**ICRH heating**

Experiments have been performed on TEXTOR in 2010 where gas has been puffed from different valves at moderate RF power, with different wall conditions and for different distances between antenna and plasma. Increase of coupling is observed when the density in the edge and in the centre is increased but oscillation of the loading resistance and changes of the phase of the reflection coefficient are also observed and could be due to global mode excitation. The fact that the gas valve was or not magnetically connected to the RF antenna does not clearly influence the change in coupling. The impact of global mode on the antenna coupling has been analysed with the help of the BRACC code. New experiments with improved plasma diagnostics will be carried out in 2011 to confirm the preliminary results obtained. Using the He beam diagnostic, density and neutral pressure will be measured in front of the antennas. Direct measurement of the phase was also implemented in the RF system and confirms the change of phase observed during the gas injection experiments.
Introduction

ITER is a large-scale magnetic fusion experiment under construction in Cadarache, France, which aims to demonstrate the scientific and technological feasibility of fusion energy production. The linear dimensions of ITER are only about a factor of 2 larger than those of the largest existing fusion experiments, while the target fusion power of 500 MW and pulse duration in the range of 10 min represent a significant step forward on the way towards an economic fusion power production.

Most of the ITER components will be provided “in kind” by the seven ITER partners Europe, Japan, USA, Russian Federation, China, Korea and India to Cadarache, with a staged approach for completion and commissioning of the ITER machine, aiming at achieving the first plasma in 2019 and full D-T plasma performance several years later. The fusion laboratories worldwide, being the carriers of key knowledge in fusion technology, are taking over significant work packages for the development of ITER components within their fields of expertise and interest, and by collaborating and transferring knowledge to industry.

Work contributions by FZJ

The institute of energy and climate research at Forschungszentrum Jülich (FZJ) is strongly involved in the development of components and methods for ITER. The main topics are as follows:

- Development of the ITER Core Charge Exchange Diagnostic system, together with the consortium partners from the Netherlands (ITER-NL), United Kingdom (CCFE) and Hungary (HAS): see the details within the “Diagnostics” and the “Technology” chapters of this report.

- Investigations on the lifetime prediction/optimisation and surface cleaning of ITER diagnostic first mirrors: see the details within the “Diagnostics” and the “Plasma-wall-interactions” chapters of this report.

- Development of methods for the in-situ measurement of Tritium retention within the first wall, material deposits and dust: see the details within the “Diagnostics” and the “Plasma-wall-interactions” chapters of this report.
• Development of a fast gas valve to mitigate plasma disruptions (DMV): see the details within the “Technology” chapter of this report.

• Divertor modelling: see the details within the “Theory” chapter of this report.

Additional work on ITER development is performed by our collaboration partners FOM and ERM/KMS, to which our institute is closely linked via the TEC agreement:

• Development of the ITER Electron Cyclotron Resonance Heating (ECRH) system, see the annual report by our TEC partner FOM (Rijnhuizen, NL)

• Development of the ITER Ion Cyclotron Resonance Heating (ICRH) system, see the annual report by our TEC partner ERM/KMS (Brussels, Belgium)
Introduction

The German Research Centres Jülich (FZJ) and Karlsruhe (KIT) as well as the Max Planck Institute of Plasma Physics in Garching (IPP) jointly develop and construct the Wendelstein 7-X stellarator in Greifswald. The stellarator concept is regarded as an attractive candidate for a future fusion reactor due to its specific potential for continuous operation. Wendelstein 7-X is a large stellarator, which has been optimized according to the quasi-symmetry principle. It consists of superconducting coils and is intended to provide plasma discharges of 30 seconds duration at a heating power of 10 MW. The aim is to demonstrate the basic suitability of the chosen concept for magnetic confinement with long pulses.

According to its existing expertise, Research Centre Jülich has taken over comprehensive work packages for the construction of the Wendelstein 7-X stellarator. This includes, above all, work for the design and manufacture of superconducting connections of the magnet system, supporting work in welding technology, structural simulations as well as diagnostics development.

Engineering

The engineering contributions of Research Centre Jülich to the construction of Wendelstein 7-X are as follows: design and construction of the superconducting busbar system including support structure, design and manufacturing of superconducting joints, and structural analysis of several components.

The Topic Technology of IEK-4 Plasma Physics deals with engineering aspects. For detailed information the reader is referred to the report of this group within its chapter addressing Wendelstein 7-X.

Diagnostics

The main diagnostic contributions of Research Centre Jülich to the construction of Wendelstein 7-X are as follows: VUV spectrometer system (HEXOS), diagnostic neutral beam injector (RuDiX), dispersion interferometer, visible spectroscopy, imaging Bragg spectrometer.
The Topic Diagnostic and Heating of IEK-4 Plasma Physics deals with diagnostic aspects. For detailed information the reader is referred to the report of this group within its chapter addressing Wendelstein 7-X.
Introduction

R&D-Programmes No.: E.11203.04, E.11205.01, E.11205.02, E.11205.03

Destructive Examination of European Divertor Vertical Target Qualification Prototypes

Three Vertical Target Qualification Prototypes (VTQP, fig. 1) mock-ups produced by ANSALDO and PLANSEE were high heat flux tested in the electron beam facility TSEFEY in St. Petersburg. After HHF testing the mock-ups were re-inspected in the SATIR test facility. In addition the two PLANSEE mock-ups were investigated by computer tomography (CT). Finally the important destructive metallographic/ceramographic examinations (light microscope) have been performed at FZJ.

Concerning the mock-up fabrication joining between the W and CFC monoblock tiles was carried out by hot isostatic pressing (HIP). A pure copper (Cu) interlayer on the CFC and W part is used between the armour and the heat sink to alleviate the thermally induced stresses in the interface (fig. 1b).

For the W part of the VTQP component the most important interest was in the field of homogeneous grain size structure in tungsten (for example elongated grain structure perpendicular to the loaded surface), no cracks and holes in tungsten, no holes in the pure Cu interlayer area and as a normal effect the typical recrystallisation in the Cu/CuCrZr cooling tube and the pure Cu interlayer.

For the CFC part of the VTQP component the interest was directed towards the field of eroded surfaces, cracks in the surfaces, significant holes in the structure, gaps between CFC and the Cu interlayer, bubbles during Cu casting in the Cu interlayer, residual Cu melt in the CFC structure, occasional binding-faults between Cu and CuCrZr, crack behaviour in CuCrZr/Cu and a different microstructure observed in the CuCrZr (fig. 1). For that reason complex micro-hardness measurements were accomplished inside the CuCrZr cooling tube cross sections (fig. 1).

Generally the micro-hardness (unit HV1) can give some information about the recrystallisation status and homogeneous or inhomogeneous stress situations inside the component after manufacturing and/or cyclic electron beam loading.
At first, micro-hardness measurements of unloaded areas in the CuCrZr cross sections were done near the CFC parts of the VTQP mock ups. These measured micro-hardness results are used as a reference. The measurements are showing that all VTQP mock ups have similar micro-hardness conditions after manufacturing. It can be recognized (as expected) that after EB loading procedures the tube wall areas recrystallised. In the upper part of the tube this effect of recrystallisation is most strongly pronounced. This range underwent the highest temperatures during cyclic electron beam loading.

It is well known that higher temperatures are correlated with a higher level of recrystallisation represented by smaller surface hardness. In the upper part of the CuCrZr tube of one particular VTQP-mock-up (fig. 1b) some ranges with higher hardness inside the low hardness range could be observed (yellow dashed range with micro-hardness values between 108 HV1 to 121 HV1). Obviously this is due to the effect of precipitation hardening.

![Fig. 1: Top view of one VTQP mock-up (a) with A, B, C = HHF units consisting of 11 CFC and 14 W blocks each. The cross-section from the CFC part (b) shows inhomogeneities of the micro-hardness in the cooling tube close to the electron beam loaded surface. The relatively high hardness values in the yellow dashed range are obviously due to precipitation hardening.](image-url)
Thermal shock resistance of different tungsten grades

The main concern about tungsten is the large variety of grades and their anisotropic microstructures which have an influence on the mechanical properties including the DBTT (ductile-to-brittle transition temperature). Furthermore, the recrystallisation temperature plays an important role in the material's performance, since recrystallisation leads to a decrease in mechanical strength, an increased ductility, and an increase of the DBTT.

In order to characterise and quantify the influence of different material compositions, microstructures, and properties on the thermal shock behaviour, several tungsten grades were investigated. All tested materials, e.g. W-UHP, pure W, WVMW, WTa1, and WTa5, were sintered and thermo-mechanically deformed in axial direction to obtain a disc-shaped geometry with 160 mm in diameter and 29 mm in height. Due to this manufacturing process, the microstructure is strongly deformed with flat grains elongated in radial direction and compressed in axial direction.

The thermal shock experiments were performed in the electron beam facility JUDITH 1. For these experiments, specimens with dimensions 12 x 12 x 5 mm³ were cut. The orientation of the specimens was defined with the loaded surface to be parallel to the front surface of the disc. The experimental parameters simulating ELM relevant conditions comprise the exposure to multiple shots (n = 100) at absorbed power densities of 0.15 – 1.3 GW/m² for 1 ms, which corresponds to a theoretical temperature increase of the material by ~280 – 2300 °C, respectively. Furthermore, the base temperature was varied between RT and 600 °C.

![Graph showing thermal shock resistance of WTa1](image)

**Fig. 2:** Thermal shock damages of WTa1 depending on the testing conditions. The green and the red dashed lines are the damage and cracking threshold respectively.
Two kinds of stresses are induced in a material during a thermal shock event. The formation of a temperature gradient leads to compressive stresses during the heating process due to the constriction of the heated area by the cold surrounding material and tensile stresses during the cooling down process. Furthermore thermal fatigue damages are induced by repetitive loading during multiple pulse tests. The inflicted damages strongly depend on the testing conditions, material properties and the manufacturing process that influences the microstructure.

An example for the induced damages depending on the testing conditions for WTa1 is shown in fig. 2.

Based on these results it was possible to define damage and cracking thresholds (see fig. 2). Below the damage threshold no damages or surface modifications occur. Above this threshold thermal shock crack networks or surface modifications like roughening can be observed at the material’s surface depending on the base temperature. For WTa1 the damage threshold is located around 0.2 GW/m². The cracking threshold was determined to be between 200–300 °C.

The characterisation of the thermal shock behaviour of several tungsten grades and the comparison of these results in combination with differences in the materials mechanical and thermal properties will lead to a better understanding of the damage mechanisms. This knowledge will help to improve the material behaviour under high thermal loads.

**Tungsten under repetitive thermal shock loading**

Type I Edge Localised Modes (ELMs) are transient events at frequencies > 1 Hz that repeatedly load the plasma facing components of a fusion device such as ITER with high heat power densities (GW/m²) for a short time span (~1 ms). Under these conditions plasma facing materials will be subject to different degradation processes (depending on material and loading conditions) like surface roughening, cracking, erosion, melting, etc. In order to investigate these processes a new experimental procedure was developed allowing loading samples with a high number of ELM-like heat loads by using a fast moving focused electron beam. The electron beam diameter is a crucial parameter, as it determines the power density of the beam. It depends on various parameters such as vacuum pressure and focussing magnetic lens’ currents. The dependency of the beam diameter on these parameters was measured, enabling tests with defined power densities.

The test specimens used are tungsten blocks (12x12x5 mm³) brazed to a copper heat sink that is actively cooled. Tests were done at three different power densities (0.14 GW/m², 0.27 GW/m² and 0.41 GW/m² at an absorption coefficient of 0.55) with different pulse numbers (10,000-1,000,000), a pulse length of 0.5 ms and a repetition rate of 25 Hz (fig. 3). The material showed no change after exposure to 0.14 GW/m² even for 250,000 pulses. This value is therefore considered as lower limit for the damage threshold. At a power density of 0.27 GW/m² a development can be observed from **no change** at 10,000 pulses to **surface roughening** at 100,000 pulses (visible to the naked eye, measured with a laser profilometer) to a **crack network** at 1,000,000 pulses. For the highest power density cracking is observed.
already after 10,000 pulses. The results not only determine the damage threshold (shaded area in Fig. 3), but also show that this threshold depends on pulse number. This means testing candidate materials cannot be finally done without high cycle tests.

Fig. 3: Results of the high cycle tests on actively cooled tungsten samples. The heat pulses have a length of 0.5 ms and are repeated with 25 Hz frequency. The shaded area represents the damage threshold.

W-coatings for the ITER like wall in JET

First wall components and materials for applications in future nuclear fusion devices need to fulfil special requirements. Especially thermal loads in magnetic confinement experiments like ITER have a severe impact on the material degradation of the plasma facing components. Tungsten coatings are being assessed for use instead of bulk tungsten grades. Within the ITER like wall project, realized at JET, a part of the thermally loaded wall will consist of 25 \( \mu \text{m} \) thick tungsten coated Carbon Fibre reinforced Carbon (CFC) modules. The coating was produced by a Combined Magnetron Sputtering and Ion Implantation (CMSII) coating technique in the National Institute for Laser, Plasma and Radiation Physics in Bucharest, Romania.

In order to quantify the material degradation under transient thermal loads (ELM-like = Edge Localized Modes) for nuclear fusion applications, small specimens of tungsten coated CFC substrates were exposed to short thermal pulses in the electron beam material test facility JU-
DITH 1 (Jülich Divertor Test Facility in Hot Cells) for various power densities and base temperatures. The damage threshold for the tested layer system of a W/Mo/W/Mo coating was found to be approximately 0.16 GW/m² for very short pulse durations of about 1 ms. This value is approximately half of the cracking threshold value compared with bulk tungsten grades. The erosion of the coating for high absorbed power densities is presented in fig. 4.

**Fig. 4**: Particle erosion of a tungsten coated CFC substrate under transient thermal loads.

The influence of the base temperature in a range between room temperature and 400 °C is negligible. Furthermore, thermal fatigue tests on tungsten coated divertor tiles for the ITER like wall project with cycle numbers up to 100,000 pulses for different power densities were performed in the electron beam test facility JUDITH 2. The results of the experiments confirm that increasing cycle numbers lead to an accumulation of damage, the so called thermal fatigue.

Finally, the failure mechanism under thermal loads was investigated by light microscopy as well as by electron microscopy. These investigations have shown that coating degradation is mainly dependent on the fibre orientation of the CFC substrate material based on the material properties like thermal expansion and heat conductivity of the single fibre orientations of the CFC substrate.
Performance of beryllium under transient thermal loads

To qualify new beryllium grades for ITER, several Russian and Chinese materials were tested in the electron beam facility JUDITH 1 and compared to the reference material S65C. In a first campaign, not actively cooled samples made from these materials were loaded in thermal shock experiments with single shots and multiple shots. The results were reported elsewhere.

In a second step, four actively cooled mock-ups were produced in Russia and in China (two of each party). These mock-ups (see fig. 5) consisted of a water-cooled CuCrZr body with four tiles from different beryllium grades. Both parties used their own joining techniques, but each of the mock-ups contained also beryllium tiles from the other party, as well as from S65C.

Each tile was 40 x 24 mm² and it was loaded by the following scenarios on different surface areas:

- VDE test at 40 MJ/m², 1 shot, a = 10 x 10 mm², 50 ms ramp-up, 165 ms steady state,
- Disruption simulation at 3 MJ/m², 1 shot, heated area a = 5 x 5 mm², Δt = 5 ms, damage factor: P·√t = 43 MW/m² × s¹/₂,
- Repetitive test with 1000 shots at 80 MW/m² (2 MJ/m²), a = 10 x 10 mm², Δt = 25 ms (this loading condition was similar to the one of the so-called Watson test which was carried out at Sandia Nat. Lab. before)

Following these loading conditions, one mock-up of each party finally underwent a thermal fatigue test with 1000 cycles at 2 MW/m², 15 s heating and 15 s cooling (heated area: whole sample surface). All experiments were carried out with water cooling of room temperature. In order to detect any detachment of tiles, the surface temperatures were controlled by IR camera before and after all experiments.

After completion of the high heat flux experiments in the electron beam facility, post-mortem examinations were performed by scanning electron microscopy (SEM) of the surfaces as well as by metallography. From these analyses no fundamental differences were found for the damage in the different beryllium grades.

Fig. 5: Actively cooled test mock-up for transient heat load tests on Russian (RF) and Chinese (CN) beryllium grades in comparison with the ITER reference material S65C (US). The large melting spots originate from the VDE-simulation test; the smaller spots are footprints of disruption simulation experiments.
Introduction

**Phenomena** at the plasma edge and their impact on global plasma behaviour remain one of the most challenging domains in fusion plasma theory. The Theory and Modelling Group of FZJ has further focused its activities towards edge plasma related issues, including plasma surface interaction, atomic, molecular processes, plasma flows and edge plasma turbulence. This group activity is covering physical topics from the entire SOL and pedestal region, “from the barrier to the target”.

Specific phenomena of interest remained:

- Recycling, pumping of fuel particles, He ash, impurities in divertor → Influence on performance
- Erosion and melting of plasma limiting surfaces, injection of neutral gases for diverse purposes and resulting local and global changes in plasma parameters
- Mechanisms of edge transport and its modification with resonant magnetic perturbations (RMP), impurity seeding, etc. with important impacts on particle/energy exhaust and modification of confinement characteristics at the plasma edge and in the core

The **Methodology** consists of

- Development of models, numerical approaches and codes of different complexity for particular phenomena
- Coupling of codes for different phenomena and plasma regions to arrive at a coherent picture
- Development of first emerging computational engineering tools, e.g. for modelling of particular ITER and DEMO design issues

Significant progress was made possible via further exploiting the dedicated European fusion high performance computer HPC-FF operated at FZJ, as well as the FZJ general purpose supercomputing facility JUROPA, in addition to further upgrading the groups own Linux Cluster with its moderate parallel technology for scaling studies.
Recycling/pumping conditions and impact on performance

Life-time estimates for diagnostic first mirrors in ITER

(ITER Service Contract C4T/09/71/OLT CHD/DIAGNOSTIC) were carried out with the linear Knudsen gas flow approximation of the Monte-Carlo code EIRENE on fixed 2D plasma backgrounds provided by ITER-IO (B2-EIRENE simulations). Realistic multi-differential spectra of neutral particles, which originate mainly from main chamber recycling, entering specific ITER diagnostic port plugs have been obtained. With these spectra 3D particle transport inside the diagnostic ducts (generic shapes) was carried out, accounting for erosion by fast atoms and deposition processes on the duct walls. It was shown that bulk erosion on Sc-Mo mirror surfaces is acceptably low in long ducts, but that deposition of impurities (Be, C …) remains an hard to quantify issue. The computations strongly indicate that a refined duct shape design with “fins” (internal apertures) can potentially reduce deposition rates by orders of magnitude. Corresponding experimental studies have been initiated to either confirm or disproof these findings.

The newly established engineering code cycle, from CAD via ANSYS (FEM grid generation) up until EIRENE 3D Monte Carlo transport (linear or non-linear), is illustrated in Fig. 1 below.
Impact of cross field transport divertor detachment was investigated in order to separate, computationally, unknown (anomalous transport) effects from classical components in the integrated computational edge plasma models. By introducing a collisionality dependent (hence also: radially dependent) cross field transport law into the JET suit of edge codes (EDGE2D-EIRENE), significant better (with respect to experimental findings at JET) characteristics of the route to detachment were achieved in the code simulations. In particular a much smoother roll over of recycling at divertor target plates with flux dropping nearly to zero, with continuously increased upstream densities was found, the ion flux roll over signatures happen already at lower plasma densities and the asymmetric transition into detachment - inner target detaches before outer one, were all now also seen in the simulations.

A coherent description of deuterium/tritium neutrals and plasma in a fusion reactor including divertor, SOL, ETB and core regions was developed, primarily to investigate overall isotope effects in a D/T burning plasma. A comprehensive numerical model for transport of charged particles with anomalous particle transport due to ITG/TE modes in the plasma core, residual anomalous, neoclassical and ELM-induced transport in ETB and with the effect of mutual thermal forces between deuterons and tritons in SOL was set up. This new model was integrated with a description for recycling neutral particles, in which atoms are described in diffusion approximation and charge-exchange between deuterons/tritons and tritium/deuterium atoms is explicitly taken into account. Regimes with strongly reduced triton recycling (and hence tritium retention) as compared to deuteron recycling fluxes, have been identified by utilizing the operational freedom provided by fuelling (pellet injection) and vacuum pumping system. It seems that far more favourable operation modes (with respect to the tritium fluxes onto divertor targets), without negatively affecting the fusion power performance, might be possible.

Erosion and melting of plasma bounding surfaces, injection of neutral gases for diverse purposes

An EIRENE module for a drift-kinetic description of weakly ionized impurity ions in the SOL is envisaged, in order to simplify existing integrated multi-codes by avoiding unnecessary work flows. The concepts of “single sourcing” (re-using well tested fluid code interfaces, A&M and geometry modules of EIRENE also for 2D/3D trace ion transport) have been followed by implementing a drift orbit solver and a Fokker Planck collision module into EIRENE (collab. with OEAW). Several tests have been carried out, and the remaining key challenges for this code integration have been identified, both by employing simple semi-analytical test models and by application to a full 2D tungsten ASDEX-Upgrade edge transport problem (in collaboration with IPP Garching). It turned out that severe challenges remain to be solved, before this much simpler integrated code (as compared to fully coupled 2D edge + trace-ion Monte Carlo codes) can be regarded as alternative option. Major difficulties arise from the use of track-length estimators (default in EIRENE) for diffusive processes in general, and the pre-
cision (or lack thereof) of explicit orbit integrators in combination with the EIRENE Monte Carlo stepping.

Modelling of (non-linear) local plasma parameter modifications induced by impurity injection, such as by methane puffs, is necessary for proper interpretation of local erosion and deposition studies, both experimental and those by linear simulation with externally prescribed background plasma fields.

Newly developed semi-analytical formulations allow studying the entire sequence of phenomena: Ionization of impurity neutrals and resulting increase of local electron density, energy losses on excitation, ionization and radiation and resulting local plasma temperature drops, both together then in turn affecting the neutral impurity penetration process.

Solving the proper spatial and temporal plasma and neutral particle balances consistently indeed results in the local plasma effects of gas sources and allows assessing the maximum gas puff source strength, below which such effects can still be safely neglected for linear erosion-deposition modelling.

**Control of edge transport with RMP**

Fast mapping approaches to model behaviour of test particles and runaway electrons in the presence of MHD instabilities have been further elaborated. Universal asymptotical behaviour of poloidal spectra of resonant magnetic perturbations (RMP) is investigated, using e.g. the I-coils topology in the DIII-D tokamak. In particular the amplitude spectra of RMP harmonics are shown to be of importance, e.g., in calculations of field lines and for characterization of their stochastic behaviour. For certain sets of saddle coils such spectra are shown to obey some asymptotic behaviour, and the coefficients arising the analytic expression for these spectra have well defined properties, e.g., they are nearly constant in the vicinity of the magnetic separatrix.

Another important application of this same mapping methodology is to formation of particle transport barriers in a small-scale chaotic magnetic field. Low-order rational magnetic surfaces are shown to play an important role for transport of particles across the magnetic field. Direct numerical simulations of trajectories of collision-less test particles in weakly chaotic fields, and comparison with quasi-linear predictions have been carried out. These calculations show that radial transport is strongly reduced near low-order rational drift surfaces for large enough Kubo numbers $= \tau_{\text{cor}} / \tau_{\text{tran}}$. Furthermore, the computed effective cross field diffusion is shown to have a fractal-like radial profile with deep transport barriers.

Calculations of 3D transport and profile modifications due to flows along perturbed field lines with fixed perpendicular transport laws with the EMC3-EIRENE code have been advanced and validated in applications to RMP scenarios relevant for TEXTOR-DED parameters. The code has been newly supplemented with a diagnostic tool to simulate synthetic Balmer radiation camera pictures for direct quantitative comparison with experimental data. Further
refinements for simulations of RMP ELM-control scenarios in H-mode plasmas with ITER similar shape have been enabled by generalization to spatially dependent cross-field transport coefficients to include effects of an edge transport barrier, and by advancement of the neutral gas pumping and gas-puffing mechanisms in the code. This work was carried out under F4E-ITER contract to assess the possible consequences of RMP to plasma surface interaction issues in ITER.

Self-consistent description of **parallel and anomalous perpendicular transport including RMP** and plasma instabilities for fixed profiles of averaged plasma parameters is provided by the ATTEMPT code. This code solves drift-fluid transport equations in configurations with magnetic field perturbations, e.g. from RMP and in the presence of plasma instabilities. ATTEMPT has been applied to clarify physics of RMP-screening features in TEXTOR-DED experiments. It was demonstrated that electric currents inside the plasma arising in response to RMP from DED generate a phase shifted magnetic field perturbation and an incomplete suppression of perturbations even in cases with full screening, $\gamma = 1$.

Further drift-fluid studies have been carried out to clarify the physics of GAM (geodesic-acoustic modes) damping observed in TEXTOR-DED experiments. It was shown that changes in magnetic field topology induced by RMP introduce a dissipation of GAM. The dispersion relation for GAM frequency taking into account RMP effects has been derived, in which the controlling parameter increases with RMP amplitude and reduces with plasma resistivity.

**Further developments of edge modelling approaches** have led to advanced numerical algorithms for plasma transport solvers, in the presence of non-monotony and even discontinuity of plasma flux as function of gradients. In particular the time evolution of electron density by instantaneous formation of the edge transport barriers can now be solved much more robustly. The novel numerical scheme is implemented into the ITM suit of core transport codes.

The edge code B2-EIRENE, currently employed for ITER divertor design, has been adapted also for linear plasma devices (collab. with K.U. Leuven) to enable code verification studies also employing such, seemingly simple plasma devices currently under construction in TEC (MAGNUM, PSI-2, JULE-PSI). The earlier work carried out during the operation of PSI-2 in Berlin (H. Kastelewicz) has been used as guidance to re-build these options and also to verify the new code applications to PSI-2 model configurations. Furthermore some newer options of B2-EIRENE, not available during the time in Berlin, have been successfully activated and tested in the linear configuration, e.g. the inclusion of parallel electric currents and correspondingly modified sheath boundary conditions. See Fig. 2 for a schematic of how, by proper re-interpretation of coordinates, the 2D tokamak edge code concept could be transferred to linear plasma configurations without need to develop any new transport code modules.
Direct application of B2-EIRENE (SOLPS) for PSI-2 is still possible, but coordinates have to be adapted.

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**MAST**: Up-down symmetric, one single separatrix, two separated SOLs

**PSI-2**
HPC-FF System Operation

The year 2010 was characterized in an effort to stabilize the system. Major problem areas were:

- Sporadic disk server hardware failures (January)
- Intermittent Infiniband (IB) hardware problems (March and July-September)

Both incidents affected the users massively as in both cases the Lustre file system got stuck and user jobs could not be finished and finally were aborted. In both cases it was very difficult to diagnose the root cause of the problems.

For the first case finally it was required to exchange the hardware for the home file systems completely as the problem could not to be identified otherwise. Later the (buggy) storage controller modules could be identified as root cause of this problem and were corrected with an updated version.

For the second case several long maintenance slots were necessary to diagnose the location of the IB errors. As comprehensive diagnose routines were not available originally it was necessary to bring all partners (Bull, SUN/Oracle, ParTec, Mellanox) together (several times) to develop an understanding of these problems and to find a way to improve this situation in the quite complex JUROPA/HPC-FF environment.

The Lustre file system contributed to this situation as this specific version (1.8.1.1) was most vulnerable to any small hardware problems in the network or storage environment. When we upgraded the Lustre version to 1.8.4 - after availability in the third quarter - we finally were quite a bit relieved from Lustre hangs as this new version seems to have built in other mechanisms to deal with these errors and so the stability of the system could be improved afterwards.

Another source of problems has been the repair actions of the hardware companies. In several cases we experienced system hangs when (pretended repaired) nodes were brought back into the production environment. It turned out that some of these nodes overwhelmed the IB network and in turn brought down the whole system. To avoid this situation we decided originally that repaired nodes could rejoin the system in a maintenance slot only to be able to check out the system before we would allow any user jobs to run. But this extended the duration of these slots even more. At the same time we began to develop health check scripts to find problems during boot time of nodes and also during user job initialization. As soon as these scripts were
ready and stable we seem now (end of 2010) to be able to reduce the maintenance slots. The number and length of these slots was the most mentioned concern of the users during this year. Nevertheless it should be noted that the JUROPA/HPC-FF complex was capable of hosting very large applications with excellent performance which could not be executed on any other system. Please see the article about “The Millennium-XXL Project: Simulating the galaxy Population of the dark Energy Universes” in INSIDE Vol. 8 No. 2 which could serve as an outstanding example.

**ITM-Gateway Connection**

The so-called S0-scenario (please see the report from the IWG = Inter-Operation Working Group from December 2009) was for the most part implemented on HPC-FF in 2010. This scenario allows ITM Gateway users to copy their data on the HPC-FF front-end (or login) nodes from their ITM-Gateway AFS-based home directories into their HPC-FF Lustre directories. Additionally the ITM-Gateway repositories and software tools will be mirrored from AFS into the HPC-FF Lustre file system once a day; in this way they become available to all (batch and interactive) HPC-FF nodes.

For the implementation the following tasks had to be fulfilled:

- The Kerberos authentication and authorization on HPC-FF
- User credentials are mirrored from the ITM-Gateway database
- Setting up an AFS replication server on HPC-FF
- The JSC LDAP server and user management was worked on to comply with ITM standards. This part will be finished early in 2011

Additionally a new hardware server had to be installed to host the new services. In summary this was much more work for ITM-Gateway and JSC system administrators as originally planned.

**Lustre Storage extension**

Early in 2010 it became clear that the installed file system capacity and performance would not be sufficient in the very near future. Originally there were ~700 TeraByte installed (~300 TB for the Scratch/Work and ~400 TB for the Home file system). Also the promised and (during system acceptance) tested performance could not be reached in real life as the Write Cache Feature could not be enabled on the DDN S2A 9900 to ensure file system protection in case of storage hardware failures.

After realizing that the most space was occupied by JUROPA users and after completing a market survey JSC decided in July to replace the old DDNs and to install new storage hardware, in summary: 4 DDN SFA 10.000:
• This extended the available space for Scratch/Work and Home to about 1.7 PetaByte
• The performance for the new Scratch/Work will now be \( \sim 17 \) GB/sec (factor \( \sim 5 \))
• The aggregated performance for all Home file systems was also raised by \( \sim 17 \) GB/sec

The installation and acceptance of the new storage was completed in December. As users have to move parts of their data (from Scratch/Work only) to the new file system, it will take until February 2011 to complete the storage extension. Later in 2011 some Home file systems will also be migrated to the new storage areas without as much interruption for the users as possible.

The standard default capacity for each project is now: 3 TB within the Home file system and 2 million files.

**HPC-FF Exploitation**

After localizing and solving a problem with the device drivers of the 10 GigaBit Ethernet adapters of the gateway nodes we were able in June

• to connect the gateway nodes to the fast DEISA network
• to enable all users to access the GPFS file systems and thus
• to enable all users to access the Archive file system

Some users having access to one of the DEISA partner computers were happy to use the fast DEISA network to transfer their data back and forth to JUROPA/HPC-FF.

The compute node utilization of both JUROPA and HPC-FF during workdays is now generally over 90%. Especially on the HPC-FF side we still see some drops on weekends and holidays thus bringing down the total average processor load to about 70%. One reason for this could be that developing work is still taking place for the majority of the fusion codes. Another sign for this assumption is that it was requested from the High Level Support Team (HLST) to increase the number of interactive compute nodes (especially for debugging parallel programs) to 64.

Also some users seem still to be afraid to submit low priority jobs when their monthly CPU quota is exhausted although they were encouraged to do so in several emails.

The cycle-swap mechanism was used by projects FSNEULA and FSNGYRT.
Test limiter experiments in TEXTOR

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Melt layer behaviour of tungsten in a strong magnetic field was studied by exposing tungsten thin plates (2 mm thickness) with gaps. We exposed four different W plates such as highly purified W (6N), W-Ta (5%) alloy, and two He plasma exposed W. Two temperature conditions were employed at 1300 K (nano-structure was formed) and 800 K (a He bubble layer was formed beneath the surface). He ion fluence to these samples was about $10^{24}$ m⁻² with an ion energy of about 150 eV. On the high temperature sample, nano-structure of about 500 nm in height was formed. These W plates were installed on a newly fabricated graphite roof limiter. The experiments were performed in the TEXTOR NBI discharges. By placing the limiter at around r ~ 46 cm, melting near the tip of the plates was observed. Melt layers on all W samples behaved similarly as they moved in the direction of JxB force. The W samples with the surfaces decorated by He plasma showed similar melt layer behaviour to the other tungsten plates. It was clearly observed that the nano-structure disappeared at around 1500 ~ 1700 K, which took place before the melting. Therefore, because of disappearance of He decorated layer before melting as well as its thinness, it can be concluded that these He induced surface morphology does not affect melt behaviour. Detailed analysis of surface morphology of molten layers and comparison with simulation works is planned.

Fine-grained tungsten with TiC dispersoids, developed by Prof. Kurishita in Tohoku University, is attracting an increasing attention because of its low embrittlement under low temperature, recrystallization, and neutron irradiation conditions. In addition, according to linear plasma experiments (PISCES), D retention was very low. But more systematic data by changing ion energy and temperature, and retention by actual confinement plasma exposure are nec-
cessary. We exposed fine-grained W to TEXTOR edge plasma for the study of retention and surface damage under actual edge plasmas. Detailed analysis such as TDS (Thermal Desorption Spectroscopy) and surface observation will be performed. In parallel, we will study retention characteristics by the ion beam device HiFIT in Osaka University.

**Tungsten material development and characterization**

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Plasma facing materials/components (PFM/PFC) will be subjected to heavy thermal loads in the steady state or transient mode combined with high energy neutron irradiation that will cause serious material degradation. Tungsten (W) is the most promising for use as PFM/PFC because of its many advantageous properties, and commercially available pure W plates in the stress relieved state are planned to be employed as the divertor material in ITER (International Thermonuclear Experimental Reactor). However, the stress-relieved W plates exhibit much lower recrystallization temperatures (1200~1300 °C) than the melting point (3410 °C) as well as coarse-grained structures with less amounts of effective sinks for radiation-induced defects. Hence, the anticipated ambience for PFM/PFC to be exposed would cause serious embrittlement to the W plates by recrystallization and radiation. Kurishita and his co-workers succeeded in significantly decreasing the ductile brittle transition temperature of W-1.1%TiC compacts in the recrystallized state by applying a new microstructural modification method based on superplasticity to ultra-fine grained (UFG) W-1.1%TiC [1, 2]: The W-1.1%TiC compacts exhibit recrystallized nanostructures with a large number of effective sinks and appreciable ductility even at room temperature, and is designated as TFGR (Toughened, Fine Grained, Recrystallized) W-1.1%TiC. Given the recent progress in materials research on W-TiC, it is appropriate to assess the alloys under closer conditions to the anticipated ambience in ITER.

J. Linke and his group at FZJ, who are world-leading in R&D of PFM/PFC in ITER and DEMO fusion reactors, and Kurishita reciprocally visited the other institute in 2007 (FZJ), 2009 (Oarai/IMR), and 2010 (FZJ) to discuss the collaboration. Kurishita’s visit to FZJ this time was also to exchange information on updated progress in the research collaboration. We are pleased to report here that it was very productive in respects that we succeeded in obtaining very noteworthy results of thermal shock tests on TFGR W-1.1%TiC together with UFG W-0.5%TiC by using JUDITH 1 (e-beam) operating at FZJ. The tests on such nanostructured W materials were the first in the world: The tests were performed by applying repetitive ELM like loads (n = 100) at various base temperatures (RT, 100 °C, 150 °C) with a pulse duration of 1
and an absorbed power density of ~1 GW/m². The main results will be oral-presented by Pintsuk at PFMC-13/FEMaS-1 (13th International Workshop on Plasma-Facing Materials and Components for Fusion Applications/1st International Conference on Fusion Energy Materials Science) on May 9-13, 2011 in Rosenheim [3] and the abstract is available online from the scientific program at the website of PFMC-13/FEMaS-1.


**Kinetic treatment of incident ions on a plasma-facing wall and its effect on the sputtering yield**


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Main source of impurity generation in a fusion device is a divertor plate and the modelling of sputtering is an important issue for impurity transport simulation. The model used in codes like ERO, however, assumes plasma without magnetic field and that can cause underestimation of the yield. In order to analyze the effect of magnetic field, PIC simulation technique was used to obtain velocity distribution function of ions in front of a plasma-facing wall. The trajectory of an ion is affected by the electric and magnetic fields and the incident angle and energy distributions have strong dependences on the angle of the magnetic field. Although the acceleration due to the electric field makes the incident angle from the surface normal small, the Lorentz force and the gyration can overcome it and the mean incident angle can become large. The larger incident angle causes larger physical sputtering yield. The quantitative analysis has been carried out by using a PIC simulation code and empirical sputtering model, Bohdansky-Yamamura model [W. Eckstein et al., Rep. IPP9/82, Max-Planck-Institut für Plasmaphysik (1993)] [Y. Yamamura et al., Rep. IPPJ-AM-26, Institute of Plasma Physics, Nagoya University (1983)].

PIC simulations of deuterium plasma with inclined magnetic field were carried out for TEXTOR plasma parameters. Statistical information of incident ions was analyzed and then the sputtering yield of deuterium on carbon was calculated from the kinetic information and the
sputtering model. The results are shown in Fig. 1. The enhancing effect of the yield is significant for large magnetic field angle measured from the surface normal, especially when the angle is larger than 75 degree. The enhancement factor reaches four in the case of nearly parallel magnetic field. On the other hand, the dependence of the yield is very weak when the angle is less than 60 degree. The parameter \( \rho_i/\lambda_{De} \) is the ratio of the mean ion Larmor radius to the Debye length and stands for inverse of normalized magnetic field strength. Although the effect of the parameter is not significant, small \( \rho_i/\lambda_{De} \), i.e. strong magnetic field, causes larger yield because of the strong effect on the particle trajectories.

This study assumes ideal plane without any roughness. The averaging effect of the roughness is a future issue. The analysis for carbon impurities in the plasma is another future issue.

**Tritium accumulation in tungsten exposed to TEXTOR plasmas**

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Tungsten (W) is one of the candidate plasma-facing materials for future fusion reactors due to its high melting temperature, low sputtering yield and low hydrogen solubility. As a plasma-facing material, W will be subjected to low-energy deuterium-tritium plasmas containing carbon impurities. In this study, we have investigated the tritium accumulation on W exposed to TEXTOR plasmas by using imagine plate (IP) technique.

Two tungsten rods, unexposed and exposed to TEXTOR plasmas, were provided by V. Philipp (FZJ, Jülich). Specimens, 5 mm in thickness, were cut from the rods so that the surfaces of the specimen represent the surface of the rods. The specimens were clean in an ultrasonic acetone bath and put then into a tritium exposed apparatus. The specimens were annealed at 573 K for 3 h in vacuum of 10^-6 Pa, and then loaded with tritium from D-T mixed gas at pressure of 1.2 kPa at 573 K for 3 h. Tritium concentration along the specimen surfaces was measured by the IP technique.

Figure shows the plasma-exposed tungsten rod mounted on the TEXTOR limiter (a), and corresponding specimens with IP images after tritium exposure (b). For comparison, the specimens from unexposed W rod and corresponding IP images after tritium exposure are given in panel (c). Digits near the IP images indicate an intensity of the photo stimulated luminescence (PSL) in PSL mm^-2 h^-1.
According to the IP images, tritium was retained in the narrow central part of the plasma-exposed rod surface. We suggest the relatively high T concentration in this area is related to a carbon deposition layer which acts as tritium trapping sites. It should be noted that the rod surface was subjected to high flux plasma exposure, and therefore carbon impurities were transported by the plasma. Tritium amount in this deposition layer is about 40 times higher than that on other surface area.

Future tasks will be performed to reveal mechanisms responsible for tritium accumulation on plasma-exposed W: Tritium loading of W (i) exposed to the TEXTOR plasmas at various conditions and (ii) W exposed to D plasma in linear plasma generator (FOM, JAEA Rokkasho, etc.) and determination of accumulated tritium by the IP technique.

Figure Tungsten rod mounted on limiter (a), and corresponding specimens with IP image after tritium exposer (b). The specimen cut from unexposed W rod and corresponding IP Image (c). Numbers in the figure are PSL intensities (PSL mm⁻¹ h⁻¹) of given position.

Excitation and suppression of Alfven eigenmodes (AEs) by using DED coil

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Alfven eigenmodes (AEs) were excited by applying small rf current (< 4 A) on DED coils. The effects of the edge magnetic perturbation on AEs have been studied by superimposing the DC current on the DED coils and preliminary results on suppression of AEs by DED fields were demonstrated. We also observed that NBI fast particle induced TAEs were suppressed by applying DED fields (3/1). In this period, we performed the identification of the externally excited AEs (TAE) for various plasma parameters (Ip, Bt and ne) and they reasonably agreed with the theoretical prediction [Fig.1].
Periscope experiment and research of dispersion interferometer on TEXTOR

Tsuyoshi Akiyama (National Institute for Fusion Science)
H. Dreier and A. Litnovsky (Forschungszentrum Jülich)

Followings are objectives of this collaboration.
1. Join to the periscope experiment
2. Research of a dispersion interferometer
3. 3rd Workshop on Interferometry for Steady State Fusion Devices

1: We have been developing passive method to suppress degradation of the reflectivity of the invessel mirror due to impurity deposition on LHD. We proposed the protective cylinder with fins. On the other hand, experiments of active method of deposition mitigation have been carried out on TEXTOR. The mirror temperature is increased and deuterium (D) gas is supplied for enhancement of the chemical sputtering of carbon deposition. In order to learn effectiveness of the active method and possibility to introduce to LHD experiments, we joined in the comparative experiments between D and helium (He) to confirm whether the chemical sputtering is the key process of this deposition mitigation. A pre-coated mirror of carbon with a thickness of 100 nm was attached in the periscope and was inserted in the SOL region of TEXTOR. The mirror temperature was kept at about 400 °C and D and He were puffed. The carbon film was cleaned only in case of D puffing. Since He does not have chemical reaction with carbon, this result indicated that the mitigation of carbon deposition is attributed to the chemical sput-
tering by D. Carbon deposition on mirror is also found in LHD and it significantly decreases the reflectivity at the visible region. Enhancement of the chemical sputtering by hydrogen puffing at mirrors during a glow discharge cleaning will be an effective way to increase the lifetime of diagnostic mirrors in LHD.

2-3: A dispersion interferometer is one of the candidates of the density measurement on future fusion device. This is because it is free from the mechanical vibrations and does not suffer from fringe jump errors by choosing an appropriate wavelength. The dispersion interferometer using a photoelastic modulator is bench-tested at NIFS. Since the dispersion interferometer on TEXTOR has been already operated and provides density data, we joined the experiments and discussed points of developing. The temperature of nonlinear crystal, which is used to generate the second harmonics of the laser light, is water-controlled within ±0.1 °C on TEXTOR while ours is only air-cooled. Since change in the crystal temperature leads to degradation of the second harmonics generation, temperature control of may be necessary for a long time measurement (~ a day) even on LHD. The phase resolution on TEXTOR is 2-3 times better than ours. This is because that back talk to the laser makes laser oscillation unstable and our back talk seems to be stronger than that on TEXTOR. After going back to Japan, the optical system is modified to reduce the back talk and the phase resolution is improved by a factor of 3. At the 3rd Workshop on Interferometry for Steady State Fusion Devices, we showed the new phase extraction method using a ratio of modulation amplitude for the dispersion interferometer and its effectiveness of the system. We discussed differences between systems on TEXTOR and ours. While the system on TEXTOR has an advantage of fast measurement ~ several µs, ours does not need developments of any special electrical circuits. The photoelastic modulator, which we use, is thermally stable for a long time. Hence, it is found that optical system of TEXTOR and ours are suitable for a fluctuation measurement and a density monitor, respectively.

Three-dimensional MHD equilibrium with resonant magnetic perturbed fields in tokamaks

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In recent tokamak experiments, it is noted that stochastic field lines reduce strong heat load driven by the edge localize mode (ELM) to the divertor plate. Stochastic field lines are produced by the external perturbed field and it is called the Resonant Magnetic Perturbation (RMP). From the viewpoint of high-β stellarator equilibrium, 3D effects on the stochastic field are very important because pressure-induced perturbed field makes further stochasticity in the peripheral region. However, in present analysis of RMP fields, a vacuum helical perturbed field superimposed on a 2D MHD equilibrium, a so-called vacuum approximation, is widely used.
Since the vacuum approximation does not include the plasma response, considerations including the plasma response and its impact are critical and urgent issue.

An important plasma response is the shielding effect of the external field. If the plasma rotates, the RMP field is shielded by the plasma response. Modelling and simulation of the shielding are done in the vacuum approximation. However, if the RMP field is superposed and the magnetic field is stochastized, the pressure and current density profiles might be changed. Therefore, the plasma boundary and geometry might be changed.

In this study, as a first step of 3D MHD modelling, the fully 3D MHD equilibrium of non-axisymmetric tokamak is solved numerically and equilibrium responses are studied. For this study, we use a 3D MHD equilibrium code HINT2, which is widely used to analyze the 3D equilibrium of helical system plasmas. Since HINT2 uses the real coordinate system, HINT2 can treat magnetic island and stochastic field in the computational domain.

3D MHD equilibrium of an ITER plasma, which includes the effect of the TF ripple and TBM, was studied. For the scenario 2, m/n = 1/1 island appeared by the perturbation of the TBM. Comparing the vacuum approximation, we found the difference of the edge structure between the 3D MHD equilibrium and vacuum approximation.
Lifetime of PFCs, Material Migration, Tritium Retention and Dust in Plasmas

The plasma conditions and the operational regimes in ITER (baseline scenario: both legs semi-detached and applied impurity seeding) with detached and seeded plasmas are challenging, and we still need a deeper understanding of the involved processes to ensure the integrity of all plasma-facing components. This is in particular critical if transients cannot be avoided, such as ELMs and disruptions. Many processes like erosion of the ITER materials C, W and Be in combination with material migration, multi-step transport, layer formation, thermal stability, gaps and melt layer behavior have to be addressed.

TEXTOR has been used to continue the investigations on local carbon migration along plasma-wetted surfaces and to remote areas such as gaps. A well proven technique to study the deposition pattern is the local injection of $^{13}$C marked hydrocarbons combined with post-mortem surface analysis. A specific comparison experiment has been performed to study in particular the deposition of higher hydrocarbons on W and C surfaces. This experiment is supposed to simulate the combined use of W and C inside the ITER divertor. Previously, the surface roughness has been identified as key player in the local deposition pattern. This effect could be avoided by polishing the surfaces of test limiters made from W and C. With these identical surface conditions the formation of local deposition layers from injected $^{13}$C marked ethane and methane on W and C surfaces could be studied. Higher hydrocarbons lead to a higher deposition efficiency. The deposition efficiency on C is a factor of 2 higher than on W and their absolute values are in the order of 1–2%. Recent injection experiments with WF$_6$ under similar conditions have shown deposition efficiencies also in the order of a few percent. These results provide important information concerning mixed material systems and will be helpful for further predictions on erosion, deposition and tritium retention in ITER.

An important tool for predictions on erosion and deposition is the numerical code ERO. The ERO code has been further developed and is used now by various partners in the European Union (Tekes, Ciemat, CEA) and also outside Europe (Japan, China). The results have been benchmarked against the EDDY code, a similar modeling code developed in Japan. Successful benchmarking of the code has been performed with data on penetration depths of CH and C$_2$ determined from spectroscopy and with the measurement of deposition patterns on W and C.

The ERO code has also been applied to linear plasmas such as PISCES and Pilot-PSI. At PISCES the behavior of the Be flux distribution to the target and the impact of Be on W or C was of particular interest. Material mixing as well as the mitigation of C erosion by the formation of BeC are of outmost interest for predictions for ITER. Investigations at these linear machines either with specific ITER-relevant material or with ITER-relevant fluxes are com-
plementary to large scale experiments in tokamaks. In particular, the ITER-like wall project at JET with the full W divertor outer target developed in Jülich will provide the possibility to study mixed systems (Be-W-C) on a large scale.

In view of the ITER-like Wall at JET, dedicated Tungsten qualification studies have been performed in TEXTOR as high risk experiments using special W tiles beyond melting. Here, the Tungsten melt layer motion and the material's splashing which has recently been observed in TEXTOR at 30 MW/m² power load have been of particular interest. The Tungsten melt layer formation and its motion over castellated structures was investigated in joint experiments on TEXTOR, addressing the issue of W melting and bridging in the ITER divertor under high heat loads. No such bridging has been observed when ITER-relevant gap distances were exposed to plasmas in TEXTOR. Melt layer motion modeling will now be benchmarked by these new experiments and predictions for ITER will gain a higher confidence level.

Already in 2009 a series of joint experiments were carried out at TEXTOR in collaboration with Japanese scientists. The investigations focused on studying the effects of plasma exposure on the prospective material concepts of the first wall of next generation fusion devices. In particular, castellated structures were used for the entire divertor armor and the first wall in ITER. However, fuel accumulation and impurity deposition in the gaps of castellated structures may be a safety issue for ITER. Modeling of the fuel accumulation inside gap structures is in the focus of present modeling activities with the new 3D GAP code developed at Forschungszentrum Jülich as a standalone solver.

Recently, linear plasma simulator experiments showed that surface bubbles and cotton-like nanostructures on Tungsten, so-called "W-fuzz", were formed by the exposure to Helium plasmas. These morphologies are vulnerable to transient heat pulses because of their reduced thermal conductivity. In addition, these structures could enhance Tungsten erosion and thus become the source of Tungsten dust. To test the survivability and evolution of such structures in a tokamak environment, pre-formed W-fuzz from NAGDIS was exposed to Helium plasmas in the scrape-off layer of TEXTOR. Here, fluences have been used that are suitable for further grow, but with higher impact energies and fluxes. The W-fuzz was completely removed in the erosion zone area of the roof limiter where the prepared tiles had been embedded: This is good news for ITER, as the W-fuzz cannot survive under relevant erosion conditions.

**Resonant Magnetic Perturbations (RMP)**

The experiments with the Dynamic Ergodic Divertor (DED) at TEXTOR are aiming at the control of transient power and particle fluxes to first wall components by applying RMP. Also, the study of power exhaust in helical divertor structures is of importance for the preparation of long and steady-state operation in stellarators resp. heliotrons.

In 2010 the DED was operated with two different electromagnetic scenarios: in the m/n = 3/1 base mode with DC and AC fields of frequencies between 1 and 5 kHz, and in the m/n = 6/2 base mode with a helical divertor topology. To tackle important questions of the RMP field penetration into the fusion plasma, the DED research program made extensive use of TEXTOR's capability to impose torque in co- and counter-current direction as well as on the capability to rotate the external field with high Ac frequencies. Plasmas with Helium as the
dominant operational gas were applied and studied reliably in direct comparison to a subsequent He campaign at DIII-D.

The recycling and exhaust characteristics of a helical divertor structure were studied in detail in m/n = 6/2 base mode. Experiments aimed at achievement of a detached plasma state in this complex 3D plasma structure. Evidence for a non-linear recycling regime were found experimentally with an increase of the MARFE onset threshold compared to a limiter discharge. The MARFE was stabilized depending on the actual magnetic topology generated by fine tuning of the RMP spectral properties and the edge safety factor value. For detailed spectroscopic studies in these interesting 3D exhaust regimes, a new spectroscopic camera system was applied allowing for quantitative analysis in comparison to EMC3-EIRENE neutral gas and plasma transport modeling. These comparisons revealed that while a non-linear recycling was found in the experiment this state could not be resembled with this sophisticate code. This points out crucial physics mechanisms at the transition to a detached plasma regime are still missing in the code when applied to high density, high recycling regimes. The potential extensions to the code will be important for thorough description of the ITER divertor plasma with this 3D code package.

Plasma edge electron density and temperature fields were studied during application of fast rotating RMP field in m/n=3/1 base mode. A characteristic modulation to the electron density $n_e(r; t)$ and temperature $T_e(r; t)$ fields in the plasma edge ($r = a = 0.9 - 1.05$) is imposed, which depends on the relative rotation $f_{\text{rel}}$ between external RMP field and plasma fluid [Stoschus et al., Phys. of Plasmas 17 (2010), 1]. It was demonstrated that the $n_e(r; t)$ and $T_e(r; t)$ modulations match the local magnetic topology as modeled in vacuum approximation for $f_{\text{rel}} = -200$ Hz. In agreement with the vacuum approximation, profile reactions of $n_e(r)$, $T_e(r)$ and $p_e(r)$ gives experimental evidence that a strongly ergodized remnant island dominated by diffusive transport exists. However, for increased relative rotation of $f_{\text{rel}} = +1800$ Hz the modulation is shifted by $2\pi$ toroidally. A pronounced flattening in $T_e(r)$ and a reduction in $n_e(r)$ suggests a shifted magnetic island. A coincident shift of the Laminar Zone as a helical SOL is observed. These first measurements of a direct dependence of the plasma edge structure on the RMP spectral characteristics and the relative rotation level point out the importance for understanding the RMP induced edge transport and the affected field penetration even in highly resistive domains. This is typically found close to the separatrix in divertor plasmas where field penetration and the effect on the edge transport and eventually the ELM stability is determined.

The strong collaboration between General Atomics and TEXTOR, having been established within this Implementing Agreement in 2006 and also conducted under the ITPA umbrella within the task "Plasmas Edge Physics 19", resulted in a significant and new physical understanding of high priority topics of the application of RMP fields for ELM control to ITER. These joint experiments focused on the following major topics:

- Investigation of plasma edge transport including identification and comparison of the perturbed magnetic topologies at DIII-D and TEXTOR.
- Benchmark of the EMC3-EIRENE code on experimental data from TEXTOR (circular, high field side limited L-mode plasmas) and DIII-D (poloidally diverted, L- and H-mode plasmas)
• Target heat and particle fluxes analysis to establish the 3D nature of the plasma-wall interaction and the SOL in the case of a future use of RMP in ITER.

• Study the resulting 3D plasma surface interaction and judge on beneficial or adverse effects during RMP ELM control by development of the coupled 3D fluid transport code EMC3-Eirene and the plasma surface interaction code ERO.

Plasma fluid and neutral transport modeling with the EMC3-EIRENE code based on the vacuum paradigm was accomplished after a development time of two years. It revealed in detail the reflection of the three-dimensional magnetic topology in the plasma structure and the non-turbulent particle flows. The structure of the perturbed scrape-off layer, including a mixture of correlated magnetic flux tubes and stochastic field lines, has been studied using this new numerical tool. First detailed reproductive profile studies have shown that it is important to include a radial dependence of the perpendicular transport coefficients in combination with a realistic reflection of neutral particle sources and sinks (external pumping system as well as wall pumping) to reproduce the profiles. This benchmark of the code against experimental data within this collaborative efforts provides a unique source for improvement towards the application of the code for ITER.

Detailed experimental investigations of RMP effects on particle transport and on changes in the global particle balance were carried out in high purity Helium plasmas at TEXTOR, and subsequently also at DIII-D. Initial analysis provided coherent evidence for a detailed control of the exhaust properties, i.e. the control of the external pumping efficiency without wall pumping by the perturbed magnetic boundary. These analysis attempts indicate that the 3D magnetic topology induced determines the actual coupling to the pumping system and hence the resulting exhaust capabilities. Continuing experiments were identified form these effort and proposed on both devices.

In detail analysis of the chemical erosion and physical sputtering in the 3D plasma boundary as seen reliably during Elm suppression at DIII-D have been carried out by the TEXTOR experimental team at DIII-D. The TEXTOR fast infra-red camera was applied for these experiments as well as numerical models for interpretation of spectroscopic systems acquired. It was shown that in a representative L-mode plasma the chemical erosion is reduced inside of the helical separatrix lobes while physical sputtering is improved due to higher energetic ions arriving at the target plates. This change in the relative erosion contributions is an important ingredient to determine the necessity of rotating the RMP field for ELM control at ITER to smear potential adverse effects for the local integrity of the target plates. Subsequent experiments at TEXTOR have made use of the two limiter locks to probe the 3D topology at two different toroidal and poloidal positions. This study qualitatively reproduced the results at DIII-D and resolved that in the 3D helical SOL imposed by the DED a SOL flow dominates the migration of eroded particles replacing the typical ExB dominated transport of impurities. These generic mechanisms resolved are essential information for the development of a combined ERO and EMC3-Eirene code package to extrapolate such findings from present day devices to RMP ELM control at ITER.
Forschungszentrum Jülich (FZJ)

Forschungszentrum Jülich (FZJ) is one of the 17 members of the Helmholtz Association of German research centres.

Structure

- Forschungszentrum Jülich is led by a board of directors consisting of in total four persons: the Chairman, the Vice Chairman and two Divisional Directors.
- Research activities are organised in four main areas, led by two Divisional Directors as members of the board: Energy and Environment, Health, Information Technology and Key Technologies for Tomorrow. "Energy and Environment" is led by Prof. Harald Bolt.

Fusion research is being carried out by the Institute of Energy and Climate Research (IEK), into which the former Institute for Plasma Physics (IPP), the Institute for Materials and Processes in Energy Systems (IWV) and the Nuclear Fusion Project (KFS) have been integrated. These research activities are being supported by the Central Technology Division (ZAT), the Operations Management Department (B) with its Hot Cell Facilities and the Jülich Supercomputing Centre (JSC) hosting the HPC-FF supercomputer for European fusion research.

Employees

- 4608 in total, comprising 1530 scientists, thereof about 549 PhD students and fellows, 1611 technical staff and 86 trainees in more than 20 professions.
- Visiting scientists: 865 from 48 countries.

Annual budget of FZJ including special tasks

- About 532 million Euro (figures with respect to 2009).

Trilateral Euregio Cluster (TEC)

The four EURATOM associated institutes for plasma physics in the Euregio (ERM/KMS Brussels, SCK-CEN Mol, FOM Rijnhuizen and Forschungszentrum Jülich) carry out a joint research programme in the framework of the Trilateral Euregio Cluster (TEC). TEC is being
steered by the TEC Directors, namely A.J.H. Donné, D. Reiter (acting), U. Samm, V. Massaut and M. van Schoor.

**Fusion Programme of Forschungszentrum Jülich (FZJ)**

The Institute of Energy and Climate Research – Nuclear Fusion Project (IEK-KFS) is responsible for the co-ordination of the fusion programme of Forschungszentrum Jülich, its leader heading the Research Unit according to the Contract of Association.

The Institute of Energy and Climate Research – Plasma Physics (IEK-4) is fully devoted to fusion research and represents about 75% of the overall FZJ fusion programme with special emphasis on the physics programme. The technologically oriented part of the FZJ fusion programme is mainly carried out in the Institute of Energy and Climate Research – Microstructure and Properties of Materials (IEK-2) under support by the Operations Management Department (B) with its Hot Cell Facilities. The Jülich Supercomputing Centre (JSC) is hosting the first dedicated European Supercomputer for Fusion HPC-FF. Further technical support is given mainly by the Central Technology Division (ZAT).

**Relevant figures of the overall FZJ Fusion Programme**

**Financial resources (in million €)**

- Consumables 1.8
- Investments 1.4

**Staffing (in PY including support by central departments)**

- Scientists 44.0
- PhD students 12.0
- Technical and administrative support 64.0
- Sum 120.0

**Visiting scientists yearly about**

100.0

**Estimated costs of the fusion programme (in million €)**

- Physics 15.0
- Technology 3.8
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