The last TEXTOR shot

Association EURATOM – FZJ

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Introduction

The Helmholtz Association's (HGF) fusion activities are in line with 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). This report presents results having been achieved by FZJ during the final year of the Association (2013).

Forschungszentrum Jülich coordinates its fusion research activities within several institutes and is well embedded into the European fusion research structures, the work programme of which is oriented along the European Road Map for the Realization of Fusion Electricity.

The major part of the Jülich research activities is located within the Institute of Energy and Climate Research (IEK). This is organized along a number of institute parts, among which fusion research is concentrated within IEK - Plasma Physics and IEK - Microstructure and Properties of Materials.

The IEK- Plasma Physics has the largest share of scientific staff in physics and technology for fusion, operated 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 development of ITER. The appointment in 2012 of a new second director at IEK-4 (Prof. Linsmeier) strengthens in particular the materials science expertise within the Jülich fusion programme. This will complement the activities which are based on the operation of the high heat flux test facilities JUDITH 1 and 2. They are installed inside a Hot Cell and in a controlled area which are licensed for operating with toxic and radiating materials.

The Association EURATOM-FZJ has very close contacts to the neighbouring EURATOM associations in Belgium and The Netherlands. In 1996 together they have founded the Trilateral Euregio Cluster (TEC) which provides a clustering of resources in order to perform a coordinated 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 supported by an IEA Implementing Agreement on "Plasma-Wall Interaction in TEXTOR" together with Japan, USA and Canada. In view of the fact that
TEXTOR has been closed at the end of 2013 and that new opportunities are arising from new linear plasma test facilities, the parties of the Implementing Agreement have agreed upon a change of name and scope: “Implementing Agreement on the Development and Research on Plasma Wall Interaction Facilities for Fusion Reactors”. This change will involve a number of existing and planned linear plasma devices in Europe, Japan and USA.

With the ending of the Association EURATOM-FZJ also the TEXTOR experiment closed operations in December 2013.

**Objectives and incorporation into the research area**

Fusion research at FZJ is to a large extent scientifically organised along topical groups and projects. This report follows this scheme covering the topical groups and projects:

- Plasma-Wall Interactions in Tokamaks
- Materials and Components under High Heat Loads
- Tokamak Physics
- Wendelstein 7-X
- Diagnostics for ITER
- Theory and Modelling
- DEMO - the route to a power plant.

TEXTOR and JET have been the main facilities for the studies of Plasma-Wall Interactions in Tokamaks and Tokamak Physics. On JET scientists from Jülich are strongly involved, in particular in the scientific exploitation of the new ITER-like wall as well as in experiments addressing ELM-mitigation. FZJ operated the TEXTOR tokamak as a local facility in Jülich ($I_{\text{p,max}} = 0.8 \, \text{MA}, B_{\text{T,max}} = 3.0 \, \text{T}, R = 1.75 \, \text{m}, a = 0.46 \, \text{m}$, plasma volume $7 \, \text{m}^3$, circular cross section, toroidal graphite belt-limiter (pumped), 16 TF coils, pulse length 12 s; auxiliary heating power: NBI co 2 MW, NBI counter 2 MW and ICRH 4 MW).

The Dynamic Ergodic Divertor (DED) on TEXTOR provided 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 participated 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.

Two air-lock systems on TEXTOR provided a backbone for Plasma-Wall Interaction (PWI) studies. They allowed exposing movable and easily exchangeable larger scale wall components. The samples could be equipped with gas feed, external heating and active cooling under ITER-relevant parallel heat and particle flux densities.

As part of the re-orientation of the scientific program after shutdown of TEXTOR, the experimental means for research on Materials and Components under High Heat Loads are enhanced significantly. Already existing facilities are the high heat flux test facilities JUDITH 1 and JUDITH 2 (ITER- and DEMO-relevant quasi-stationary heat fluxes on $50 \times 50 \, \text{cm}^2$) based
on electron beams and MARION, a 60 keV, 6 MW hydrogen/helium beam (40x32 cm\(^2\) from 10 ms to 30 s duration). We operate also the linear plasma device PSI-2 which allows PWI research under steady state conditions. The most significant investment is a new linear steady state plasma device (JULE-PSI) inside a Hot Cell in the same building as JUDITH1/2.

Fundamental aspects of erosion and hydrogen retention in ITER-related mixed materials will be studied in dedicated laboratory experiments using accelerator-based techniques (e.g. RBS and NRA) and surface analysis techniques (e.g. XPS and TPD). New experimental facilities are created at FZJ and dedicated preparation equipment is made available at the synchrotron HZB-BESSY II, Berlin. A strong new focus is established at FZJ, beginning in 2013, in the field of advanced materials for plasma-facing components for DEMO and a fusion reactor.

FZJ contributes to the scientific exploitation of **Wendelstein 7-X** based on the existing expertise in Jülich in the field of plasma-wall interactions. The general goal is to understand and qualify the island divertor including a coherent assessment of the long-term integrity of the first wall and divertor. A suite of diagnostics is considered. Particular emphasis is put on diagnostics for edge transport studies for investigating particle and heat exhaust in the island divertor. These edge diagnostics are accompanied by diagnostics for studying the core-edge interface transport and impurity levels. Those systems, i.e. HEXOS VUV spectrometer and X-ray spectrometer have already been delivered to and installed at Wendelstein 7-X.

Based on significant special funding by the German government, FZJ is working on R&D and prototype development related to **Diagnostics for ITER**. The focus is on the ITER core charge exchange diagnostics (CXRS) which will be provided by a European consortium coordinated by FZJ. A corresponding framework partnership agreement with Fusion for Energy (F4E) has been signed and the first grants are expected for this year.

**Theory and Modelling** is an important part of the PWI programme. Apart from continued direct close collaboration with ITER IO, JET and EFDA associates, on integrated edge transport codes, such as B2-EIRENE (ITER), and EDG2D-EIRENE (JET), SOLED2D-EIRENE (CEA) the specific focus is mainly on studying erosion/deposition processes with a focus on T-retention by co-deposition under ITER-like material mix conditions and the particular geometrical boundary conditions (e.g. gaps). Plasmas with pronounced 3-d topology are addressed with the code package EMC3-EIRENE code (carried out jointly with IPP Greifswald) and 3D equilibrium including resistive current redistribution by HINT2 as well as self-consistent 4-field drift-fluid plasma response modelling by ATTEMPT.

The European **DEMO** working group has the aim to develop a conceptual design for a next step device relevant for the time after ITER (PPPT project of EUROfusion). FZJ is involved in this project within the main fields of expertise, in particular concerning the materials development, plasma-surface interaction studies, alternative target and first wall concepts and divertor modelling and system studies. The German fusion laboratories IPP, KIT and FZJ have entered into a close collaboration on DEMO related issues within the “German DEMO working group”, which is organised within 13 topical groups.
One of the most challenging issues in fusion research in view of a steady-state power plant is the understanding and control of Plasma-Wall Interactions (PWI). At first, in a reactor erosion processes due to the impinging plasma flux to the wall are leading to impurity production which must be fully compatible with a sufficiently low core impurity concentration to achieve sustainable burning plasma. Secondly, the lifetime of wall components is governed by erosion, deposition and mixing processes of wall material under steady-state and transient conditions. Finally, the tokamak safety with respect to the long-term tritium inventory is determined by co-deposition and tritium implantation in plasma-facing components. Moreover, long-term tritium retention is critical for the tritium cycle balancing the tritium production and tritium breeding in the reactor. The present PWI tokamak activities in FZJ are addressing the urgent PWI-questions of ITER and DEMO on the road to a reactor as defined by the EFDA roadmap including the ITER Physics Task Forces on PWI and the JET Task Forces as well as the International Tokamak Physics Activity (ITPA) under the umbrella of the ITER organization.

In particular, the JET ITER-Like Wall provides the only integrated test bed to study PWI processes in the material mix in ITER: a full tungsten divertor and a beryllium main chamber. FZJ participated in JET work programme in form of the task force leadership, scientific coordinators of dedicated experiments and the scientific expertise in the area of PWI and power and particle exhaust. The key contributions listed below are related to work performed in this context.

**Scientific Exploitation of the JET ITER-Like Wall**

PWI has changed significantly after the removal of all carbon based components in JET (JET-C), such that the plasma with this ITER-like wall (JET-ILW) is now facing only components made of beryllium or tungsten. The replacement of C by Be induces changes in (i) the impurity source and material migration, (ii) in the recycling and retention, (iii) in the radiation and impurity content as well as impacts the (vi) plasma performance and (v) plasma operation. The use of C masked in JET-C connections between PWI and the confined plasma and reveals the bare plasma physics,

(i) Reduction of the residual carbon content. The carbon influx at the edge plasma decreased by a factor of 20. The main chamber source is reduced by a factor 4 due to absence of thermally activated chemical erosion with B without energetic threshold. The material migration from the main chamber into the divertor (x7) and in particular to the remote areas is drastically reduced (x50) with respect to JET-C. Be is not performing a multi-step transport like C, but sticks in average after 1 or 2 steps whereas C performed 10 or more steps.
The source strength of Be was determined in dedicated limiter experiments and benchmarked with the ERO code, revealing an overestimation of the primary erosion in the code by a factor. The code was applied for ITER Be lifetime predictions.

(ii) The long-term deuterium retention is reduced by a factor 10-20 depending on the plasma conditions. This was measured by gas balances, confirmed by post-mortem analysis and modelled with the WallDYN and ERO code. The main mechanism for the retention are co-deposition with beryllium in main chamber and divertor as well as implantation in Be and W. In contrast to the long-term retention, short-term remains high and is determined by supersaturation of Be. Outgassing gains relative importance and impacts, via recycling at the beginning of a discharge, on the steadiness of plasmas. As benefit of the emptied wall after a discharge or a disruption is the break-down with the JET-ILW always successful and well controlled. WallDYN reproduces the migration pattern and confirms co-deposition as main cause for retention. Transfer to ITER and multiple ITER reference plasma conditions provides 3000 to 12000 discharges without the requirement of active cleaning due to the inventory limit.

(iii) High plasma purity in the absence of C is observed (C_{Be}=1.1%, C_C=0.1%), and the JET-ILW provides conditions reflecting almost the bare deuterium plasma. Moreover, the divertor radiation is reduced due to the absence of carbon radiation and the low radiation potential of beryllium. Recovery by isotope exchange by plasma operation, ICWC or glow discharge cleaning will be required earlier to reprocess tritium for plasma operation.

(iv) Due to the absence of significant intrinsic impurity radiation in the divertor, less impurities and the low radiation potential of Be, is the operational window at higher densities extended. This leads to higher divertor electron temperatures and the expansion of the density limit up 30% slightly above the Greenwald-fraction of 1. The operational window with full detachment in both legs and stable MARFE formation is wider with respect to the upstream density and allows stable and controllable operation as required in ITER. The density limit occurs at the same radiated fraction like in JET-C confirming that density limit was masked by a radiation collapse in the carbon device. Studies in H-mode showed a similar beneficial increase of the density limit in H-mode (back transition to L-mode), but the absence of the amount of injected neutral beam power.

(v) The radiation level during JET-ILW disruptions is much lower in the absence of the intrinsic carbon radiator in JET-C. As a consequence the current quench rates are reduced and a significant amount of magnetic energy is being dissipated in form of heat flux to the main chamber wall. This led to a significant increase of the peak temperatures at the walls during disruptions and occasional melting of Be PFC at the top of the vessel during vertical displacement events. Massive gas injection with a valves developed in FZJ employing a mixture of Ar/D_2 is applied to provide the necessary radiator in a short time of less than 5ms. Successful mitigation of disrup-
tions and associated heat loads has been demonstrated for a large set of plasma conditions and plasma currents as function of the amount of injected Ar in the mixture. These finding are impacting on the design and number of mitigation valves in ITER in order to ensure the integrity of the ITER main chamber.

The use of the W divertor reduces the JET operational window inhibiting operation with sole deuterium fuelling from the deuterium stored in the wall. A scheme to control the W in the plasma was developed. FZJ was leading the part minimising the W-source in the divertor by gas injection of strong impurity seeding.

ELM control at JET by applying resonant magnetic perturbations (RMP) has been demonstrated with both, JET-C and JET-ILW. Depending on the collisionality, an increase in the ELM frequency of up to a factor of 20 has been found with moderate impact on the plasma confinement. Analysis of stability boundaries has shown that RMP application at JET induces edge safety factor resonant increase of the ELM frequency due to a shift of the stability limits due to peeling modes. This contrasts with the standard view on RMP ELM control, where mainly changes in pressure gradients, i.e. ballooning limits, are considered.

**Scientific Exploitation of TEXTOR**

### Carbon erosion and transport in TEXTOR

The key TEXTOR results can be described as follows: the resulting local $^{13}$C deposition efficiency is (i) higher for spherical limiters with a grazing angle of incidence of the magnetic field lines comparing to the roof-like geometry, (ii) higher on the graphite than on the tungsten substrate due to the different mass, (iii) higher in Ohmic than NBI heated discharges due to the difference of impact energy, (iv) increases when the limiter is biased to rise the incident ion energy, and (v) becomes higher when decreasing the puffing rate and less plasma perturbation. The $^{13}$C deposition efficiency is higher for $^{13}$C$_2$H$_4$ than $^{13}$CH$_4$ for comparable discharges.

The last experiment in TEXTOR has been applied to study the long range material migration within the all-C device. $^{13}$CH$_4$ injection through a test limiter was and about ten times more molecules than in previous experiments were injected in 118 plasma pulses to accumulate an amount of $^{13}$C in the whole vessel sufficiently high for the post-mortem analysis. After the experiment, a selection of tiles from the main toroidal limiter were retrieved for the investigation of the toroidal distribution of $^{13}$C. The local $^{13}$C deposition efficiency on the test limiter was found to be a factor of $\sim$10 higher than in the previous experiments with a lower amount of injection. A ratio of $^{13}$C/$^{12}$C of $\sim$10% was detected on a remote collector probe, well above the natural abundance. The depth profiling of deposited layers revealed that constant ratios of $^{13}$C/$^{12}$C were achieved, indicating steady state conditions for the long range carbon transport.

### Fuel removal of a-C:D layers by Ion Cyclotron Wall Conditioning (ICWC)

Ion cyclotron wall conditioning has been extensively tested in the TEXTOR device in order to optimise the fuel and carbon layer removal rate in view of the initially discussed carbon di-
ICWC has the advantage to be able to operate quasi-steady state with the magnetic field switched on. Moreover, in contrast to plasma cleaning discharges no reabsorption of co-deposits at the contact zones occurs.

Quasi steady-state operation of ICWC in multi pulse mode (0.5 s pulse every 20 s) for a maximum of 1 hour was applied for the employing the toroidal field coils in CW mode at $B_t=0.23T$ and coupling power in the high harmonics scheme. Values for the carbon erosion rate were obtained by exposing different laboratory carbon layer samples and pre-exposed TEXTOR tiles installed in the limiter lock systems of TEXTOR. The removal rates in $O_2$ and $D_2$ amount to:

- Removal rate for deuterium at 350°C was 84 nm/min of $5 \times 10^{17} \text{ D/cm}^2\text{min}$
- Removal rate for oxygen at 350°C was 38 nm/min or $2.4 \times 10^{17} \text{ D/cm}^2\text{min}$

The hundred times higher flux of deuterium to the samples appears to overcompensate the ten times higher effect erosion yield of oxygen. Thus, the removal of a-C:T by ICWC in deuterium is promising without compromising the main wall with oxygen. There is necessity to improve the ICR plasma production rates in oxygen to make it suitable for the application in a reactor.
Investigation of plasma facing materials under synergistic loading conditions in fusion reactors

During 2013, a development has been started at FZJ to investigate synergistic effects which govern the performance of plasma facing materials under fusion relevant loading conditions, comprising extreme head loads, both stationary and transient, high plasma flux densities at high fluencies and neutron irradiation causing substantial damage within the plasma facing materials.

- (Transient) heat load tests: e-beams (JUDITH), laser irradiation, including very high cycle numbers to assess fatigue effects.
- Plasma exposure: high flux densities/ high fluencies in linear plasma devices (PSI-2, JULE-PSI).
- Capabilities to investigate neutron irradiated materials and components.

These facilities will finally be combined in the Hot Material Laboratory at FZJ with its planned extensions and supplemented by additional devices outside the controlled area facilities.

Refurbishment of the Hot Materials Laboratory (HML)

The Hot Materials Laboratory (HML) of Forschungszentrum Juelich was established in the 1960s. After a refurbishment concerning the building and the supply units of the laboratories the planning for the concept of a new PSI-lab inside the Hot Materials Laboratory was completed in 2013. The main installation is a new linear plasma device (JULE-PSI) inside a new Hot Cell, equipped with a target analysis and exchange chamber to load neutron activated material samples and to perform in-vacuo characterization of surface conditions and hydrogen content after exposure with laser based surface diagnostics (Laser induced ablation and desorption, laser induced break down spectroscopy).

JULE-PSI will be assembled and tested outside of the HML. Tendering of components has recently been started. Additionally, outside of the Hot Cell but inside the controlled area of HML, a combined thermal desorption (TDS) and laser induced desorption (LID) device will be installed, which is also currently being tested outside of the controlled area.

Technical developments for JULE-PSI at the linear plasma experiment PSI-2

The optimization of the plasma source for JULE-PSI is currently being carried out on the linear plasma experiment PSI-2 outside the controlled area of HML.
To overcome the shortcomings of the cathode of the plasma source in PSI-2, which has due to its cylindrical geometry a hollow and thus inhomogeneous plasma profile, a new plasma source is under development following the set-up used at the linear plasma device PISCES-B and the University of California, San Diego, USA.

**Investigations of plasma-material-interactions in the linear plasma device PSI-2**

The targets are exposed to the PSI-2 plasma by means of a manipulator system which allows a linear motion and rotation along and tilting across the magnetic field axis. This gives the highest flexibility for the orientation of the probe surface with respect to the diagnostic applied in the plasma and analysis chamber. Further features are an actively cooled target holder, the allocation of 5 thermocouples and 3 electrical connectors, and a gas supply. The whole manipulator is electrically insulated against the vessel ground to perform bias experiments with voltages up to 300 V. The PSI-2 target manipulator has been used to expose various materials to both plasma and laser-induced loads.

**Fuel retention in tungsten**

Tungsten is to be used as plasma-facing material for the ITER divertor due to its favourable thermal properties, low erosion and fuel retention. Bombardment of tungsten by low energy ions of hydrogen isotopes, fluxes of which will vary by several orders of magnitude at various locations along the divertor, can lead to surface modifications and influence the fuel accumulation in the material. In addition to the particle flux, these changes are strongly affected by the surface temperature during the exposure.

Hydrogen retention in fusion reactors can be significantly influenced by the presence of plasma impurities. Earlier studies showed that helium can reduce the retention in tungsten wall materials. Experiments to investigate the impact of impurities on fuel retention in tungsten have been conducted in the linear plasma generator PSI-2. Exposures of polycrystalline, polished tungsten samples to a deuterium plasma were performed at low temperature (380 K) under the variation of the impurity type (He, Ar), concentration (0-5 %), and total deuterium fluence (2\cdot10^{24} m^{-2} - 2\cdot10^{26} m^{-2}, only for He), with subsequent investigations of the surface morphology and deuterium retention. The results show a reduction of the deuterium retention by a factor of 3 for helium, and no reduction for argon. Comparing the helium results with a diffusion model, the results can be explained by a shallow layer of porous helium nano-bubbles in the top surface.

**Liquid metals as plasma facing materials- exposure in TEXTOR**

Liquid metals have been proposed as alternative materials for plasma facing components in magnetic fusion devices to improve the lifetime of the components by replenishment of the liquid and to avoid neutron induced damage in the plasma facing material. However, MHD stability of the liquid, particularly during plasma disruptions, is a major issue for the application of liquid metals. To overcome this problem, so-called capillary porous systems (CPS) have been developed where the capillary pressure on the liquid, which is embedded into a solid met-
al mesh, provides stability against MHD forces. So far, CPS were exclusively tested using Lithium as liquid metal. However, the high vapor pressure of Li, its high fuel uptake and possible safety issues, pose questions on the use of Li as plasma facing material in a fusion reactor.

At the tokamak TEXTOR, we investigated the behavior of the same system during plasma exposures and induced disruptions. A first experiment was performed with a CPS preheated for longer times (≈12h) in TEXTOR vacuum to T ~ 300°C and then exposed 2 cm behind the LCFS. The plasma conditions where plasma current I_p=350 kA, toroidal magnetic field B_T=2.25 T, additional heating power by neutral beam injection, P_{NBI} =1MW yielding plasma conditions at the LCFS with electron temperature of T_e=65 eV, electron densities n_e=3x10^{18} m^{-3} and ion energies of E_{ion}= 325 eV). In this experiment expulsion of very fine droplets was observed in each plasma shot but no additional considerable mass loss was observed during disruptions. The total mass loss was measured to 5x10^{-3} g of Sn including the observed droplets. During a second exposure at B_T=2.5T, heating of the CPS was induced during plasma operation by exposing the CPS to the SOL. In this experiment no significant droplet emission was observed from the CPS surface but the better wetting capability of the Mo holder led to Sn migration and accumulation on the Mo clamping leading to droplet emission.

During exposure of the CPS to higher heat fluxes (by moving the sample to the last close flux surface), maximum surface temperatures of ≈1170 K where reached, leading also to a crack of the Mo clamping by which the Sn escaped and covered the whole holder surfaces at its edge. 0.2g Sn had left the actual CPS structure which must be attributed mainly to the material redistribution onto the Mo holder.

**Characterization of materials and components under high heat loads**

**Investigation of fusion relevant heat loads and particle fluxes on tungsten**

Double forged tungsten (as received and recrystallized state) with a purity of 99.97 wt% and single forged ultra-high purity tungsten (purity 99.9999 wt%) manufactured by Plansee AG were exposed to fusion relevant transient heat loads and particle fluxes at room temperature and 400 °C. The experiments were performed in the electron beam JUDITH 1 and PSI-2 with a pulse duration of 1 ms, a total number of 100 pulses, absorbed power densities between 0.19 and 0.96 GW/m² and a repetition frequency of 0.5 Hz to ensure a cool down of the samples between pulses. Test with combined hydrogen plasma (flux: 2.5 - 4.0 × 10^{21} m^{-2} s^{-1}) and transient heat loads were performed with the same pulse duration and repetition frequency but with 1000 pulses and absorbed power densities of 0.3 MW/m² and 0.58 MW/m². After the experiments the induced surface modifications on all samples were investigated by laser profilometry, optical microscopy and scanning electron microscopy (SEM). In addition the cross sections were examined by metallography to analyze the crack propagation into the bulk material. After experiments at room temperature without plasma background the surfaces show no damage or change in surface morphology at power densities of 0.19 GW/m². Surface modifications like roughening as well as crack networks are induced on the loaded area for both simulation methods for power densities of 0.38 GW/m² and higher.
The observed crack networks show differences in pattern of the cracked area as a consequence of the varying size/shape of the exposed region and the kind of load. With an increase of the power density the thermally induced stresses rise, cracks expand and surface roughness increases (cf. figure 1). The analysis of the crack parameters (distance, depth and width) showed that all three parameters increase with power density for both simulation methods. By comparing the power density dependence of the arithmetic mean roughness $R_a$ some differences between the two simulation methods were determined, namely for electron beam exposed samples the roughness increases very fast while for the laser beam the increase is less steep. Additional tests with combined deuterium plasma and transient heat loads simulated by laser beam were performed on ultra-high purity tungsten. Figure 2 shows light microscope images of the induced surface modifications and damages.

**Fig. 2:** Light microscope images of the induced damages after simultaneous exposure. a) exposure to thermal shock events ($0.3 \text{ GWm}^2$) and subsequently to deuterium plasma; b) simultaneous exposure to thermal shock events ($0.3 \text{ GWm}^2$) and deuterium plasma; c) exposure to deuterium plasma and subsequently to thermal shock events ($0.3 \text{ GWm}^2$); d) exposure to deuterium plasma and subsequently to thermal shock events ($0.58 \text{ GWm}^2$).
The results show that the damage behavior strongly depends on the sequence of particle and heat flux exposure, indicating that pre-exposure with hydrogen plasma deteriorates the thermal shock behavior significantly. Crack networks and surface modifications are noticed for the pre-exposed samples (figure 2c and d), whereas loading with hydrogen plasma after the thermal shock events shows no visible surface damage/modification (figure 2a) and simultaneous exposure only initiates very pronounced surface roughness as well as small cracks but no crack network formation (figure 2b). An explanation for the observed crack networks in the case of pre-exposed samples could be hydrogen embrittlement and/or a higher defect concentration in the surface near region due to supersaturation of hydrogen. For a better understanding of the underlying mechanisms further experiments with different plasma parameters (flux, fluence, particle energy, sample base temperature etc.) are necessary.

**Tungsten under high pulse number transient high heat flux loads**

Type I Edge Localized 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 focusing magnetic lenses currents. The dependency of the beam diameter on these parameters was measured, enabling tests with defined power densities.

![Image](image-url)

*Fig. 3: Overview of damages induced by pulsed heat fluxes of different number and power density on actively cooled mock-ups (example shown on top) most of them tested in the JUDITH 2 electron beam facility. The grain orientation of the tungsten tiles is shown schematically (longitudinal).*
The test specimens used are tungsten blocks (12×12×5 mm³) brazed to a copper heat sink that is actively cooled. Tests were done at different power densities (0.14 – 0.55 GW/m² at an absorption coefficient of 0.55) with different pulse numbers (10³ – 10⁶), different base temperatures (200 – 1200 °C), a pulse length of 0.5 ms and a repetition rate of 25 Hz.

The resulting surface deterioration was categorized and the damage mapping is depicted in figure 3. It shows the case of 700 °C base temperature and a tungsten grade of industrial purity with elongated grain structure aligned parallel to the loaded surface. It was found that the damage threshold (below which no surface change can be detected) is located between 0.14 GW/m² and 0.27 GW/m². At 0.27 GW/m² a slow development could be observed from surface roughening at 1,000/10,000 pulses (visible to the naked eye, measured with a laser profilometer) to small disjunct cracks at 100,000 pulses and to a crack network at 1,000,000 pulses.

For the highest power density cracking is observed already after 10 pulses. The results not only determine the damage threshold but also show that this threshold depends on pulse number. This means testing candidate materials cannot be finally done without high pulse number tests.

Experiments using the “transversal” grain orientation (perpendicular to the loaded surface) or a recrystallized material show a lower damage threshold (< 0.14 GW/m²), however especially transversal grain orientation is beneficial for preventing melt formation. The resulting cracks less likely propagate parallel to the surface and the heat removal is therefore not hindered (parallel cracks act as thermal barriers).

**Qualification of ITER first wall components**

The ITER FW components consist of beryllium tiles on the surface, high strength copper as heat sink material, and stainless steel as structural material. Advanced engineering and technologies are required to manufacture these components. High heat flux (HHF) testing represents the most appropriate methods to qualify the design and manufacturing techniques. Under grant GRT-154 with Fusion for Energy, water-cooled first wall mock-ups (MU) have been tested in the electron beam facility JUDITH2, with the aim of determining the performance of the MUs under thermal fatigue, and for qualifying the design and fabrication techniques for these components. The Be/Cu/CuCrZr/SS MUs are manufactured by using HIPping.

Dedicated calibration experiments were performed in order to fulfill the narrow parameter tolerances. These quality demands were the motivation for several experimental improvements. Amongst them is the calibration of measuring devices according to a schedule in regular intervals so that their functioning is guaranteed, and a test parameter review according to a checklist before a test campaign starts in order to minimize the risk of unperceived wrong parameter settings. Furthermore the two most valuable enhancements are the heat flux flatness verification and the electron beam monitoring.

One major criterion is the compliance of the heat flux flatness. To verify this, a new calorimetric test set-up has been developed for this particular purpose. Castellated graphite blocks with suitable geometrical dimensions, in particular with regard to the height of the single graphite...
columns that was predetermined via FEM analyses, are loaded by the electron gun. The loading is applied for a defined time while the temperature increase of the graphite surface is measured with a high resolution IR camera. The electron beam path pattern based on a digital scanning mode is optimized iteratively until the flatness requirements are fulfilled. Finally, the heat flux variation could be decreased from ±10 % to ±7 %.

U-NHF mock-up
Upgraded Normal Heat Flux (U-NHF) MUs were loaded cyclically (30 s beam on and 30 s beam off) under 2 MW/m² with ITER relevant coolant water conditions: inlet temperature 70° C, pressure 2 MPa and coolant velocity 3 m/s. A picture of the MU is shown in Figure 4.

Fig. 4: Picture of a U-NHF MU. A thin (ca.2 mm) copper layer is used to join each Be tile to the heat sink made from CuCrZr which is attached to a stainless steel back plate.

The testing conditions and MU performance are under constant supervision. Single color and two color pyrometers are installed to measure the temperature of a predetermined area of the beryllium surface. An IR camera is installed to measure the temperature distribution of the full Be surface. Thermocouples (TC) are used to measure the bulk temperature of the Be tiles. For an absorbed power density of 2 ± 0.1 MW/m², which is compliant to the testing requirement (deviation tolerance of 5%), the pyrometers indicate that the peak surface temperature of the beryllium surface is between 420 °C to 440 °C (see Figure 5). An interlock system is set up between the IR camera and electron beam gun (EBG). Once a temperature above 600 °C is detected by the IR camera the EBG switches off automatically.

Fig. 5: Temperature comparison between FEM and experimental results.

Fig. 6: Equivalent (von-Mises) stress distribution calculated by FEM.
3-D FEM analyses have been performed to benchmark the experimental results. Due to the symmetric geometry, half of the major part of the MU is used as the simulation model. The utilized tool is ANSYS. Figure 8 shows a temperature comparison between experimental and simulation results, indicating that the FEM calculated temperature has good agreement with the thermocouple measured temperatures, while the surface temperature calculated by FEM method is around 60 °C lower than the experimental data. This discrepancy is supposed to be due to variations of the emissivity of the Be surface under electron beam exposure. Figure 6 shows the FEM calculated equivalent (von-Mises) stress distribution of the MU, which indicates that high stresses are located at the interfaces of Be&CuCrZr and Be&copper; the maximum equivalent stress (588 MPa) occurs at the joining interface between tile Be2 and the CuCrZr heat sink. The high stress concentration in this area is a clear indication of the location of the failure initiation.

**Semi-prototype Mock up**

A semi-prototype (SP) ITER FW MU is being tested in the electron beam facility JUDITH-2 to determine the performance of the U-NHF design under thermal fatigue.

Before the HHF experiment, a thermal analysis is performed to evaluate the expected temperature value and distribution. In particular, the surface temperature (maximum temperature) of Be is estimated. The calculated value is used to assess the satisfaction of the criterion of test facility, which indicates the maximum Be temperature doesn’t exceed 600 °C. Figure 10 shows the 3D geometry of the SP MU. It consists of 84 Be tiles with a size of 30 mm×47 mm×10 mm. 12 thermocouples (TCs) are installed in the Be tiles for the bulk temperature measuring.

![Fig. 7: 3D geometry of the SP MU.](image)

The designed loading conditions are given with a water temperature of approximately 70° C, at 3 MPa and a flow velocity of 3 m/s. The heat flux cycle was prescribed with 30 seconds electron beam on and 30 seconds electron beam off.

The major part of the MU is adopted for the model of FEM analyses (the water inlet & outlet tubes are omitted). The temperature distribution of the MU partially loaded with 2 MW/² for
30 s is shown in Figure 8. It can be seen the maximum temperature (373° C) of the MU appears at the Be tile which is located at the center of the loading area.

![Fig. 8: Temperature distribution of the MU partially loaded with 2 MW/m² for 30 s.](image)

**Tile size mock-up**

In order to analyze the tile size effect on the performance of ITER FW component, a medium scale MU, namely tile-size MU is tested with ITER relevant coolant water conditions. A picture of the MU is shown in Figure 9. The MU contains beryllium tiles with two different dimensions, which are 141.8 mm×50 mm×10 mm and 94.2 mm×50 mm×10 mm, respectively. It loaded cyclically under various surface heat fluxes (1-2.25 MW/m²).

![Fig. 9: Tile-size mock-up.](image)
After a total of 1800 thermal cycles, no visual damage was observed in the MU. In order to analyze the temperature distribution of the MU during the testing, a 3D FEM thermal analysis was performed using ANSYS14 workbench. The maximum temperatures of MU loaded with various power densities are illustrated in Figure 10. A comparison of FEM calculated temperature and thermocouple measured temperature is shown in Figure 11. It is found that the FEM calculated result is in good agreement with the experimentally measured one.

Fig. 10: Temperature of the individual with various absorbed power densities for 30 s.

Fig. 11: Comparison of FEM calculated temperature and thermocouple measured temperature (experimental data comes from cycle 1251).
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, the effect of RMPs on plasma rotation and edge turbulence, and the interaction of RMPs with plasma instabilities. Disruption studies in 2012 aimed at the understanding of runaway electron generation and loss processes, and the exploration of massive gas injection to suppress the generation or mitigate the impact of runaway electrons. The good accessibility and a set of complementary turbulence diagnostics (electric probes, spectrometry, reflectometry) allowed detailed investigations of turbulence properties at the plasma edge.

Disruptions and Runaway Electrons

Influence of Resonant Magnetic Perturbations on the De-confinement of Runaway Electrons

Previous results from TEXTOR have shown that a resonant magnetic perturbation above a certain amplitude was able to suppress the generation of runaway electrons in discharges which were deliberately disrupted using massive gas injection. The critical perturbation field amplitude above which the suppression has been observed showed a coincidence with the field strength required to excite a locked mode in the plasma, although a test experiment with locked mode excitation didn't show the runaway suppression. In order to resolve this ambiguity a series of dedicated experiments has been performed. In these experiments the wave form of the RMP field amplitude has been carefully controlled to avoid the excitation of a locked mode prior to the thermal quench. It has been found that the application of the RMP had almost no effect on the suppression of runaway electron generation. Test experiments where a locked mode has been deliberately excited before the thermal quench showed no runaway electrons. Furthermore, it was found that the locked mode had to be excited some time before the disruption in order to efficiently suppress the runaway generation. A possible explanation for the observed behaviour could be that the runaway generation in these experiments is governed by the so-called hot tail generation mechanism. The cooling of the core plasma due to the enhanced radial transport by the locked mode results in much less hot electrons after the thermal quench which could act as a seed for the runaway electron population. Further analysis of both experi-
ments, the previous one suggesting the RMP suppression effect and the new experiment showing that there could be in addition a governing role of the hot tail mechanism, is ongoing.

**Massive Gas Injection for Runaway Electron Suppression**

The TEXTOR tokamak is equipped with two fast disruption mitigation valves. A smaller valve which can inject up to ~1 bar.l of gas has been used to deliberately trigger a radiative collapse disruption, reliably generating runaway electrons. A larger valve which can inject up to ~11 bar.l of gas has been triggered either simultaneously or with a prescribed delay. In a series of systematic scans the delay time, the gas amount, and the gas species has been varied. The obtained data allows to determine the range of parameters where the suppression of the runaway electrons occurs. The main findings from these parameter scans are:

- A developed runaway electron beam can not be suppressed by massive gas injection, i.e. in order to be efficient the second gas injection needs to be within 3-4 ms after the disruption trigger. Anyhow, some de-confinement of runaway electrons due to collisions with the injected gas is always visible and leads to a temporal decay of the runaway electron current. As long as the runaway plasma can be controlled and doesn't touch the tokamak walls this is a beneficial effect for runaway mitigation.

- Injection shortly after the thermal quench results in no runaway generation, as long as the injected amount of gas was above a critical level. In this case the runaway generation has been suppressed. Future data analysis will compare the obtained results with the theoretical prediction of the critical density for runaway suppression from Connor-Hastie and Rosenbluth.

- The critical amounts of gas depend on the species. The free electron density has always been below Connor-Hastie value, showing that in addition the bound electrons in neutral atoms and partially ionised ions contribute to the collisional slowing down of energetic electrons accelerated by the Dreicer field. The data suggests that the bound electrons contribute less to the friction force with higher Z elements.

Further analysis of the data will be done in the course of a PhD thesis.

**3D Field Physics for Transient Plasma Wall Interaction Control**

**Introduction**

An edge localized mode (ELM) crash leads to fast, repetitive losses of energy and particles from the plasma edge, resulting in peaked heat fluxes onto the plasma facing components, mainly in the divertor region. The high energy and particle fluxes may be unacceptable for large tokamaks like ITER. Non-axisymmetric perturbations (3D fields) are one option for controlling ELMs. These perturbations are applied as resonant magnetic perturbations (RMPs) and their feasibility has been demonstrated on various, different sized, devices such as MAST,
AUG, DIII-D and JET. However, the current understanding of the involved processes for ELM control is low. An open question is: what is the influence of RMPs on the dynamic of ELMs?

In this section, recent results from TEXTOR, JET and EAST on 3D field physics for transient plasma wall interaction control are summarized.

**Plasma response to 3D magnetic perturbations on TEXTOR and JET**

On TEXTOR, shielding currents have been observed to form as a plasma response to applied rotating RMP fields. Multiple shielding currents on neighbouring resonant surfaces have been observed concurrently for perturbation fields of different amplitudes, frequencies and phases. A systematic comparison of the dynamics of the plasma response to RMPs shows a good qualitative agreement between the experimental observations and the results from quasi-linear MHD modelling. The understanding gained from these TEXTOR results has been applied to JET ELM mitigation experiments, where the plasma response to applied fields with toroidal mode number \( n = 2 \) shows a clear correlation with ELM mitigation.

**On the effects of magnetic perturbations on fast ion losses studied at TEXTOR**

One criterion for the ignition of a fusion plasma is sufficient fast ion confinement. A key aspect in that context is the maintainability of good fast ion confinement in the presence of non-axisymmetric fields, such as those found in stellarators and during the application of RMPs in tokamaks.

A study of the RMP influence on fast ion losses at TEXTOR has been carried out on TEXTOR. The measurements were taken with a rotating directional probe located at the mid-plane at a fixed toroidal angle providing information along the radial axis in the SOL with a resolution \( \Delta r = 6 \) mm. From discharges without applied RMPs, the strong dependence of the losses on the location of the LCFS could be seen, and the appearance of passing particle losses could be excluded at the measurement position. During this work, clear RMP effects on the fast ion losses are only seen when investigating the edge safety factor dependence. At a low edge safety factor about \( q_a \approx 4 \), a strong reduction of the losses by 50% is observed when applying strong static perturbation fields with a poloidal/toroidal mode number 6/2. At a moderate edge safety factor \( q_a \approx 6 \), RMP rotation scans in co- and counter-direction with frequencies of up to 5kHz show no clear effects on the fast ion losses at TEXTOR. These scans have been performed for poloidal/toroidal mode numbers of 3/1 and 6/2. Theoretical studies that involve tracing the fast ions are required to understand the impact of RMPs on the losses at TEXTOR and can help towards a further interpretation of the experimental results.
**Findings of pre-ELM structures through the observation of divertor heat load patterns at JET with applied \( n = 2 \) perturbation fields**

On JET, the pre-ELM structure has been observed for the first time in the divertor heat loads. This structure is seen by the appearance of a dynamic, quasi-stable heat flux pattern on the divertor plates a few milliseconds before the ELM crash accompanied by an increase of the \( D_\alpha \) emission (see Fig. 1). The pre-ELM structures appear on a much longer time-scale than that of typical ELM crash rise times. This suggests that the phase prior to an ELM crash has an effect on the particle transport to the divertor before the ELM crash. The appearance location of the pre-ELM structures might be explained by considering thermoelectric currents in the plasma edge. Their radial propagation may be caused by rotating ballooning modes. Further studies, including full transport modelling, are needed to come to a conclusive understanding of the observed phenomena.

The pre-ELM structures are not restricted to the application of magnetic perturbations, although they are seen to be strongly enhanced and appear regularly, when the \( n = 2 \) fields are present and above a threshold similar to that causing saturation of the ELM mitigation. Observed splitting of the ELM crash heat deposition appears to be influenced by the pre-ELM structures, whether this is a general feature of the pre-ELM structures remains open. Further experiments and analysis are needed to quantify this effect.

**Modified heat load deposition of the ELM crash due to \( n = 2 \) perturbation fields at JET**

Significant changes in the edge localized mode (ELM) crash heat load deposition patterns compared to typical ELMs are seen via infra-red observations during resonant magnetic perturbation experiments at the Joint European Torus (JET). These modifications result from the changed magnetic topology of the plasma, caused by the perturbations. The experimental results show that the modified heat load deposition of the ELM crash depends on the perturbation
strength and the edge safety factor. A thermoelectric current model shows that current filaments in the plasma edge could explain the observations. This study gives an insight into how the changed magnetic topology affects the peak heat fluxes of ELMs which is crucial for understanding ELM control.

A new method of ELM control using lower hybrid waves on EAST

Recently, ELM mitigation has been observed on the Experimental Advanced Superconducting Tokamak (EAST) when lower hybrid waves (LHWs) are applied to H-mode plasmas sustained mainly with ion cyclotron resonant heating (ICRH). This has been demonstrated to be due to the formation of helical current filaments (HCFs) flowing along field lines in the scrape-off layer induced by LHWs. Because of the geometric effect of the LHW antenna, the perturbation fields induced by the HCFs are dominated by the n=1 components, where n is the toroidal mode number. In comparison to previous RMP ELM mitigation experiments, using a set of fixed in-vessel coils, ELM mitigation with LHWs on EAST is achieved with a wider range of q₉₅ as shown in Fig. 2.

For the low q₉₅ discharge, the influence of LHWs on ELMs can be rather quick in time, and ELMs can even be completely suppressed. However, a longer delay time before the appearance of large ELMs was observed in the high q₉₅ discharge after fast switching-off of LHW, sometimes longer than 50 ms, as seen in Fig. 2 c).

It is worth noting that ELM control with RMPs induced by in-vessel coils is normally limited by a narrow resonant q₉₅ window due to the fixed coil geometry. However, the HCFs induced by the LHWs flow along the magnetic field lines in the SOL, thus the helicity of the HCFs always closely fits the pitch of the edge field lines whatever the value of the plasma edge safety factor. In addition, the HCF-induced magnetic perturbations are localized at the plasma edge, without significantly affecting the plasma core. Therefore, this may provide a better method for ELM control on ITER.
Installation of the PCR system at AUG

The poloidal correlation reflectometry (PCR) at TEXTOR has been stopped operating in spring 2013. The diagnostic was used at TEXTOR for the measurement of plasma rotation and turbulence properties, based on the reflection of launched micro waves in a certain frequency from the plasma. The system undergoes a major refurbishing with the aim to operate the diagnostic at ASDEX Upgrade (AUG) from beginning 2014 for the study of shear flows at the plasma edge. This topic is embedded in the frame work of the Helmholtz virtual institute (VI) for the application of advanced microwave diagnostics in fusion plasmas.

All in-vessel components had to be designed to fulfill the in-vessel requirements at AUG. Antennae properties of the antennae array consisting of a launching and 4 receiving antennae has been calculated to allow a maximal gain of the antennae. Furthermore the antennae is designed to allow O- and X-mode polarization to cover a large radial region of typical AUG plasmas. To run both polarizations on the same wave guide fundamental wave guides cannot be used. Therefore a slightly over sized waveguide with a square cross section is chosen. A complicated waveguide path for a bundle of 5 wave guides of nearly 5m has been designed, fabricated and tested in the institute. Therefore special bending tools for the bending of the wave guide are developed and tested. The in vessel installation was ready by November 2013 at AUG (see fig. 1). In parallel the detection of the reflected waves has been modernized with new balanced mixer. A computer controlled motorized microwave switch allows the measurement of radial correlations by hooking up a 2nd reflectometer. Furthermore the existing frequency range of 24-36GHz at TEXTOR is extended by a third reflectometer covering the frequency range of 40-60GHz. The whole system was ready for operation in late December 2013.

Investigation of 3D quasi coherent structures

The reflectometry system at TEXTOR was during the last campaign used for the study of long range correlation between toroidally and poloidally separated antennae arrays, with the aim to
study the toroidal correlation length of turbulent structures. It has been found that the structures of a certain frequency range (50kHz-120kHz) are correlated along field lines, only. These structures exhibit not the classical turbulence features but a more detailed substructure as could be investigate with short range correlation within one antennae array, only. From the estimated auto correlation these structures show the behaviour of a damped oscillator with a damping ration in the order of 1. The decay constant of these structures in the order of 10μs. The period of the the oscillation is variable and has a minimum of 5μs which increases for later oscillations up to 8μs. The natural frequency of the structures in the order of 60kHz-110kHz. Because of the decaying coherent oscillations in poloidal direction the structure appears like a quasi coherent mode. Since the structure exhibits several oscillations the estimation of the propagation time from cross correlation calculations yield different propagation times for different oscillations. This feature is nicely visible in the long range correlation measurements where up to 4 oscillations can be identified. This fact that the quasi coherent modes are aligned along magnetic field lines and suggest they are also visible in magnetic fluctuations. A possible explanation for the quasi coherent modes as micro islands is under investigation. The observed high poloidal mode number could be a hint in this direction. Also possible consequences for the interaction of quasi coherent modes with the geodesic acoustic mode seems possible. Both modes have a large toroidal extent which makes the interaction probable.
Wendelstein 7-X is the largest stellarator in the world built in Greifswald, Germany, by the Max-Planck-Institute for Plasma Physics (IPP), Greifswald with considerable support by the German Research Centres Jülich (FZJ) and Karlsruhe (KIT). The mission of this helical advanced stellarator (HELIAS) type device is to show good energy and particle confinement with highly reduced net-currents in the plasma equilibrium obtained. This route promises an intrinsic stationary confinement of fusion relevant high temperature plasma.

One key ingredient for successful operation of Wendelstein 7-X is the reliable and stable exhaust of energy and particles to maintain the first wall and divertor integrity and keep the plasma clean enough for sustained fusion power production. The divertor concept used is the island divertor. It employs magnetic islands in the plasma edge, which represent the interface between the core plasma and the plasma facing components. The qualification of this island divertor concept is mandatory to explore the capability of a HELIAS type stellarator with island divertor as a candidate for a future fusion power plant.

To tackle this goal, a versatile diagnostic suite was defined and realized by FZJ within a new founded working group structure on “Plasma Surface Interaction in 3D Plasma Boundary at Wendelstein 7-X”.

**Scientific Vision of the Contribution to Wendelstein 7-X**

In order to prioritize the technological efforts when developing new diagnostic systems for Wendelstein 7-X, a scientific vision with a clear time line was developed correlated with the overall schedule of the Wendelstein 7-X project. In this overall schedule, two operational phases are foreseen. Operational phase 1 (OP1) will start in 2015. In this period the machine will feature pulsed operation of up to 50s at low power (1MW) and 10s at high power (10MW). Electron Cyclotron Resonance Heating (ECRH) will provide the heating power during this phase. The divertor will be an inertial cooled Test Divertor Unit (TDU) with mechanical pumping only, i.e. turbo pumps. During this first time frame, research will target on the exploration of generic aspects of this new and first of its kind device. For the divertor physics, qualification of the basic functionality of the island divertor as exhaust concept is the major goal. Exploration of the divertor regime for different density levels, pumping and neutral pressure capabilities and the local divertor parameters are of central interest.

Subsequent to OP1 a hardening and completion phase of 18 months will be conducted to implement the actively cooled High Heat Flux (HHF) divertor, the cryogenic pumping system and additional heating capabilities (Ion Cyclotron Resonance Heating (ICRH) and Neutral Beam Injection (NBI)). This will allow stationary operation (30 minutes) at 10MW and high
power pulses at 18MW for 10s. During this phase, the divertor physics will focus on aspects of stationary operation addressing for instance in how far the divertor can be operated in a detached regime with reduced power fluxes and to which extend the pumping efficiencies allow stationary density control. This characterization of the pulsed and steady state features of divertor operation will enable to address the long-term integrity of the divertor and first wall, for instance the erosion efficiencies and redistribution of divertor material during steady state.

The novel feature of the island divertor is the inherent three-dimensional (3D) structure of the plasma boundary and the fact that the island chains in the edge - which depend in location and size strongly on the internal plasma currents – determine the performance and characteristics of the divertor scheme. This requires in consequence local diagnostic capabilities in the edge plasma, to validate prediction on the local magnetic topology – for instance the rotational transform in the island domain. The approach developed allows addressing key aspects with regard to these high priority questions during the first two operational phases at Wendelstein 7-X.

**Diagnostic Alignment in 3D Magnetic Field Topology**

The diagnostic systems envisaged as a systematic setup to contribute significantly to the divertor exploration at Wendelstein 7-X is depicted in figure 1. This is result of a dedicated analysis of the magnetic field topology conducted in 2012.

*Fig. 1: Prioritized diagnostic systems and the systematics within the magnetic topology of Wendelstein 7-X*
Local divertor spectroscopy employing filter based camera systems allowing tomographic reconstruction of emission lines as well as high resolution spectrometric analysis is envisaged as working horse for the characterization of the divertor regime. This includes spectroscopy on intrinsic species as carbon or hydrogen as well as on actively injected gases like helium. Line ratio spectroscopy on thermal helium will be used for measurement of local profiles of electron temperature and density. To deploy this measurement technique developed at TEXTOR a versatile gas inlet system was developed. It was delivered to Wendelstein 7-X in 2013 in time for first operational phase.

For measurement of parallel profiles, a fast reciprocating probe system is planned at the upstream position, i.e. at the symmetry point of the connection between the lower and upper divertor observation locations. They are projected in figure 1 onto one toroidal plane but in the experimental realization they are 180 degree toroidally separated as shown in figure 2 showing the magnetic topology at the two toroidal planes where the diagnostics will be located and the parallel magnetic connection.

This diagnostic setup spans up one out of five divertor sets in the five-fold symmetry of the machine and allows to quantitatively assessing local divertor parameters as well as parallel dependencies in the 3D topology. The parallel flows determining the divertor regime will be established along the magnetic field lines but also from inside the island attached to the divertor plate. A tangential density measurement along the inner island surface provides an averaged density value at this local upstream position. This can be provided by the dispersion laser interferometer developed and tested at TEXTOR. It is planned to deliver the four TEXTOR channels to Wendelstein 7-X and we plan using one channel for measurement at the inner side of the local island divertor island chain as sketched in figure 1.

Characterization of the transport at the interface between the plasma core and the plasma edge is crucial to understand the interaction between the plasma and the material wall and vice versa. The fast reciprocating probe system enables to characterize the turbulent transport in the outer part of the island located at the mid-plane. This system is accompanied by correlation reflectometer. These systems together allow for quantitative measurement of key quantities like density and temperature fluctuations, correlation time and length scales for density and – derived from this – information on the magnetic field structure, i.e. the local magnetic field line pitch angle.
Progress in Development of Key Diagnostic Components

The diagnostic systems rely on development of three key components, which has been started in 2012 - the versatile gas inlet, versatile observation endoscopes and the fast probe system.

The versatile gas inlets were designed and were manufactured and delivered in 2013. An overview on this system is shown in figure 3. The gas boxes are constructed from five separate micro tubes closed by local Piezo valves, which allow separate activation of each gas inlet. A prototype of this system was tested in a laboratory experiment, and suitable low leak rates for Wendelstein 7-X vacuum conditions have been confirmed. The system can be operated up to 70 bars enabling fluxes of $10^{24}$ at/s. In this operational mode the system can be used for local divertor fueling and divertor density regime control. At low pressures the fluxes amount to $10^{18}$ at/s suitable as non-invasive active gas puff diagnostic. The pulsed system avoids in this mode accumulation of the diagnostic gases. The overall setup in figure 3 shows that this design included not only the gas box itself, but the entire vacuum flange plug.

The endoscope system is in the design phase. For the endoscope system an imaging mirror based system is envisaged with a small aperture closed by a pneumatic shutter. The system is laid out as water-cooled setup allowing application for the high heat flux phase as well. The light will be transferred to outside the vacuum with interface optics for flexible light distribution to a variety of spectroscopic observation systems. At each gas inlet position two endoscopes are considered for local tomographic reconstruction of the light emission from both, actively injected gases as well as intrinsic impurity species like carbon for instance.

The fast reciprocating probe system was specified in regard of the probe head requirements and reciprocating speeds. This is based on a physics assessment identifying that this system is envisaged being a user facility for using a variety of different diagnostic probe heads. The main priority is given to a classical triple Langmuir-probe head for measurement of stationary and turbulent time scale quantities. A Conceptual Design Review in was held in 2013. Detailed design has started. The aim is to deliver the system by end of 2014 in order to be ready for the initial operational phase OP1.
In addition to the three key diagnostics the concept of a **poloidal correlation reflectometry** system has been developed in 2013 based on experience with similar systems at TEXTOR and ASDEX Upgrade. A system of five antennae together with their wave guides is cantilevered from the cryostat flange (figure 4). This module is to be integrated into the reflectometry port of Wendelstein 7-X together with other microwave diagnostics. The frequency synthesizer has been specified as well.

**Fig. 4: Concept of the in-vessel setup for the poloidal correlation reflectometry composed of five antennae.**

### An Ion Cyclotron Heating System as Collaborative TEC Effort

FZJ and the Royal Military Academy (RMA) Brussels as two TEC associations developed in 2012 a conceptual design for an Ion Cyclotron Resonance Heating (ICRH) system for Wendelstein 7-X. This involved physics assessment of the radio frequency requirements to provide substantial heating of ions in Wendelstein 7-X. The particular aim is generation of ions in a relevant energy class of a future fusion device to benchmark the fast ion confinement.

The expertise of RMA personnel enabled to provide two in detail requirement definitions for RF heating with resonant frequencies in the range of 25-32 MHz and up to 76 MHz. The first frequency range is compatible with the TEXTOR antenna setup and generators and has been selected in the conceptual design review. The system will be able to provide up to 2 MW ICRH power for up to 10 s. The conceptual design of the present setup was matured in 2013.
6. ITER Diagnostics and Technology

Conceptual Design of a CXRS diagnostic for the core plasma of ITER

The core CXRS (Charge Exchange Recombination Spectroscopy) diagnostic is designed to measure in the core plasma of ITER not only the density of the helium produced in the fusion reactions but also other quantities which are relevant to the physics or to the control of the tokamak: ion temperatures, the density of various impurities and velocities.

The IC3 (ITER core CXRS Consortium) is an ad hoc association under the lead of FZJ as coordinator: KIT (Germany), BME (Hungary), CCFE (UK), FOM-DIFFER and TU/e (the Netherlands) are active members in the consortium, supported by third parties in Spain (CIEMAT) and Hungary (Wigner RCP, Optimal Optik Kft). Physics modelling with updated atomic data, conceptual design studies and prototype development were carried out in the frame of research programmes supported by the German BMBF. The IC3 Consortium was selected by the European Agency to ITER (F4E) for the development of the core CXRS diagnostic in the upper port plug 3 of ITER in a first phase of four years under the umbrella of a Framework Partnership Agreement.

A possible port plug design and arrangement is shown in the figure above. Electromagnetic, nuclear, thermal, and mechanical shock loads in vacuum combined with high accuracy and stability requirements; protection and cleaning mechanisms for mirror surfaces subject to erosion and deposition are all real challenges in the harsh environment of the ITER torus. Several
functional units were envisaged, with special emphasis on the protection of the first mirror. An example is given by the prototype shutter shown below.

![Recent prototype of a fast shutter to protect the front mirror in the port plug](image)

The progress achieved in development of the ITER CXRS port plug and of its components provides a sound basis for the design review to be held at ITER in 2015 as well as for the full development of the diagnostic within the awarded Framework Partnership Agreement with Fusion for Energy.
Introduction

The physical phenomena at the plasma edge of a tokamak and in a linear plasma device have a crucial impact on the global plasma behavior. This interplay includes surface interaction, atomic, molecular processes, plasma flows and edge plasma turbulence.

The Theory and Modeling Group of IEK-4 specifically focused on these edge plasma related issues and their work is covering physical topics from the entire SOL and pedestal (transport barrier) region, up to the non-linear plasma sheath at exposed surfaces. The interplay of non-linear plasma dynamics, solid state physics and atomic physics requires the development of fundamental theoretical models and the massive use of computational techniques as well.

The particular research topics are:

- Interaction of the charged particle plasma with neutral gas, impurities in divertor and at the plasma edge with emphasis on plasma performance
- Local and global changes in plasma parameters due to spreading of impurities injected for diverse purposes (radiation cooling).
- Edge transport and its modification due to resonant magnetic perturbations (RMP), with important impacts on particle/energy exhaust and confinement modification at the plasma edge and in the core plasma.
- Plasma physics in linear simulators, boundary conditions at the plasma-wall interface.

Methodological aspects are:

- Development of models, numerical approaches and numerical codes.
- Coupling of codes for different phenomena and plasma regions to obtain a consistent modelling approach covering several magnitudes of temporal and spatial scales.

Via further exploiting the dedicated European fusion high performance computer HPC-FF operated at FZJ and its successor IFERC (Japan), as well as the FZJ general purpose supercomputing facility JUROPA, significant progress in the field of numerical plasma physics was possible.
**Plasma response to local impurity sources**

A strongly localized and intense release of impurities can strongly modify the overall plasma parameters. This requires an extension of the theoretical approach beyond the trace particle approximation considering the impurities as passive with negligible retroactive effects.

A 3D shell model for spreading of impurities in the Massive Gas Injection (MGI) experiments has been further improved. The model is based on 3D fluid equations averaged over certain regions of shell cross-sections, resolved by magnetic surfaces. Such approach allows fast calculation of time variations for radial profiles of the parameters of the main plasma and impurities. Entire sequence of phenomena which take place on the thermal quenching phase of MGI is described: ionization of injected neutrals and resulting increase of local electron density, energy losses on inelastic processes, local electric fields, the effect of cooled down plasma on the penetration of neutral impurities.

**2D transport modeling with B2-EIRENE**

Sophisticated computer codes are inevitable tools to investigate the complex interplay of plasma dynamics, neutral particle physics and plasma-wall interaction. The B2-EIRENE code, developed at FZJ, is a widely used tool for detailed edge plasma simulations and tokamak design studies as well. The code became part of several projects and has been permanently improved and adapted for a variety of applications.

The B2-EIRENE code has been adopted for 2D simulations of plasma in the linear devices, such as PSI-2, JULE-PSI and MAGNUM. The goal of this activity carried out together with TEC partner KU Leuven is to contribute to model validation by employing such, seemingly simple, plasma devices. B2-EIRENE models for the plasma simulator PSI-2 have been set up and tested, comparison with experimental measurements has been started.

Development of the new version of the B2.5-EIRENE code (SOLPS-ITER) has been completed. The new code package developed under F4E service contract in collaboration with St. Petersburg Polytechnic Univ. and KU Leuven has to replace SOLPS4.3 as the principal ITER edge modeling code. The package consists of the version of B2.5 plasma code SOLPS5.2 with the most up-to-date model for classical drifts and currents coupled with most recent version of EIRENE (fully MPI parallelized, with non-linear Monte Carlo for neutral-neutral collisions, radiation transport and photo-excitation). Conformance of the new modeling code to the ITER convergence and accuracy criteria has been demonstrated, the solutions are verified against the SOLPS4.3 results.

**Accuracy and performance of CFD-MC combination (B2-EIRENE)**

Most sophisticated edge modeling codes to date readily available for practical calculations – such as B2-EIRENE or EDGE2D-EIRENE – combine fluid (finite-volume Computational Fluid Dynamics - CFD) model for plasma coupled with kinetic Monte-Carlo (MC) model for neutral particles. For such a CFD-MC combination no convergence criteria exist which would be generally accepted in numerical mathematics. In turn, stopping criteria for the iteration
scheme and accepted level of residual error have strong impact on the wall-clock time of computations.

In order to make the first step in addressing this issue the CFD residuals have been generalized for the CFD-MC combination. On each time-iteration of the time-marching relaxation scheme B2-EIRENE applies internal iterations in order to reduce CFD residuals after the MC call. This approach, however, leads to severe restriction of the time-step, thus, to very large amount of time-iterations required for steady-state solution – the wall-clock time of one simulation for the realistic ITER or DEMO scenarios reaches several months. The time-step can be increased by (at least) two orders of magnitude if no internal iterations are used, but it was found that in this case generalized residuals saturate on much higher level, and, in particular, violation of the global particle balance drives the solution into completely wrong direction. A correction scheme has been developed and tested which keeps accurate particle balance while still allowing large time-steps.

With this correction, large CFD residuals of equations other than continuity equations remain an issue. Two approaches have been tried to resolve this problem: i) correlation sampling in the MC part which makes two MC runs made on close plasma background nearly identical; ii) extrapolation of the neutral solution to the changing plasma without invoking the full MC simulation – this is advancement of the “EIRENE short run” technique used originally in B2-EIRENE in the beginning of 90s. The current status is that there exist modeling cases where those techniques can reduce CFD residuals down to round-off error, but the methods do not work for full scale ITER models with strong impurity radiation and detached divertor.

The problem of CFD-MC coupling itself has a fundamental nature and is not related directly to the particular implementation in the B2-EIRENE code. Therefore, convergence analysis of the B2-EIRENE runs was complemented by a study of strongly reduced “academic” test cases done in cooperation with KU Leuven. This study indicates that the discrepancy in time scales between slow cross field transport of neutrals from target to the pump and fast parallel electron heat transport increases with increased machine size, rendering the problem numerically increasingly stiff. At the same time the non-linearity increases in the coupling between kinetic (neutral) species and fluid (bulk plasma) components, rendering the integrated code system increasingly more vulnerable to MC noise.

3D transport modeling of edge plasmas with EMC3-EIRENE code

Several phenomena at the plasma edge need a detailed consideration of the full three-dimensional dynamics, e.g. in the presence of perturbed magnetic fields and/or stellarator configurations.

Numerical investigation of the implications of an ITER RMP system on divertor characteristics with EMC3-EIRENE made under F4E contract has been completed. The model was further refined to include the spatially dependent cross-field transport coefficients - edge transport barrier. In ITER simulations - performed for the reference Q=10 H-mode
scenario - a strong density pump out is seen inside the separatrix. A high sensitivity of the results with respect to assumptions on RMP screening is identified.

**Stratified source sampling in EMC3-EIRENE** has been implemented. Superposition of several independent neutral particle sources, as it is already routinely in place in B2-EIRENE, allows improved statistics in the neutral-related terms, modelling of gas puffing at the edge, as well as volumetric recombination. Realistic model of gas puffing/pumping is applied in DIII-D and ITER simulations. Inclusion of volumetric recombination in the model will allow EMC3-EIRENE to access detached modes of operation with cold divertor plasma.

**Plasma edge modeling for Wendelstein 7-X with EMC3-EIRENE** has been carried out for the reference scenario based on a vacuum magnetic field configuration. The transport of intrinsic carbon impurities was modeled as well as the transport of injected helium and nitrogen. Predictions of the impurity line radiation are made in support of the diagnostic development at FZJ. Simulations have been performed for the limiter plasma start-up scenarios in order to estimate the heat loads on the plasma facing components. Modeling of the local nitrogen seeding showed that radiation cooling controlled by the gas inlet system developed at FZJ can be used for wall protection.

**Drift-fluid modeling with ATTEMPT code**

The non-linear interplay of electric fields and currents on the micro-scale (ion and electron Larmor radii) gives rise to the so-called anomalous transport on the macro-scale (centimeters up to device dimensions). To take into account the micro physics properly a drift-fluid approach is employed to be used in the local approximation (fast and small scales only) to provide transport coefficients for transport codes like B2 or EMC3, or on the basis of the full global approach to capture both, micro and macro scales.

The ATTEMPT code, developed at FZJ, provides time-dependent solutions of the 3D drift-fluid transport equations resulting in self consistent description of the parallel and (anomalous) perpendicular plasma transport.

The **parallelization of the ATTEMPT code** led to a two orders of magnitude speed-up in terms of wall-clock time for typical applications. To obtain this increase in computational performance the classical non-local algorithm for solving the Helmholtz-equation has been replaced by an algorithm which allows an efficient distribution of floating-point operations on a parallel platform. The parallelized ATTEMPT code was embedded into the data structure of the EU Integrated Tokamak Modeling project on the GATEWAY computer. The code structure has been adapted to provide a turbulence module conformable with the CPO (Consistent Physical Object) data transfer rules and is prepared for routine applications in ETS KEPLER workflows.

For **studies on 3D plasma dynamics in linear devices** the ATTEMPT code has been extended to a global approach, based on a fully self-consistent treatment of electric fields, drifts, electric currents and sheath physics and describing the evolution of the full plasma profiles. Strong
impact of the upstream plasma source (e.g. hollow vs. flat cathode) on the intermittent ("blobby") transport characteristics is found. The plasma conditions are mostly influenced radial gradient region in the source profile (as, e.g., given by hollow cathodes as opposed to flat cathodes). The statistical properties of the drift-fluid turbulence show clear similarity to what is observed in tokamaks as well (experimentally and theoretically), i.e. a pronounced parabolic relation between the kurtosis and skewness parameters of the plasma density fluctuation time series. A second important effect found in the plasma dynamics of linear devices is the ion-neutral friction separating a stable and smooth plasma operation from strongly turbulent regimes. Therefore, the recycling and other kinetic effects (e.g. on the basis of EIRENE), shall be considered in connection with such a more detailed plasma model.

**Coupling of local drift-fluid turbulence to plasma edge transport codes** is investigated to provide a more solid physical basis for cross-field transport coefficients for the edge codes, which are currently free model parameters. Based on flux-driven ATTEMPT runs, a scale separation of the plasma dynamics in the area of closed flux surfaces of the plasma edge can be detected. The coupled code system, consisting of the gradient driven (local) ATTEMPT code for the micro-scale dynamics and a 1D code for macro-scale transport, has been compared to flux driven ATTEMPT simulations, capturing both the micro- and macro-scale evolution. Good accordance is found regarding the time evolution of flux surface averaged radial density, with time savings for the coupled code system of at least one order of magnitude. The multi-scale modeling approach is currently expanded towards the more complex 2D edge code B2.

**Test particle transport in RMP fields**

For several phenomena Hamiltonian mapping techniques offer the possibility for half-analytical and computationally less demanding numerical algorithms compared to sophisticated but time-consuming codes.

**Fast Hamiltonian mapping approaches to model behavior of test particles**, in particular runaway electrons in the presence of MHD instabilities and small-scale magnetic turbulence has been developed. Test particle transport in the weak turbulent wave-fields at large Kubo numbers is shown to have fractal structure. Strong reduction of radial transport near low-order rational drift surfaces is found. Collisional transport of test particles in stochastic magnetic fields has been modeled numerically by employing a combination of concepts from Hamiltonian chaos theory and random (diffusive) kicks. A semi-empirical formula for the effective cross field diffusivity has been derived from numerical experiments. Calculations predict reduced cross field particle transport at increased plasma density in agreement with experimental observation made in the tokamak DIII-D.

**Computational engineering**

**Adjoint computational divertor engineering/optimization techniques** are further elaborated in cooperation with KU Leuven and RWTH Aachen. The concept of “pde constrained optimization” known from the aerospace and automotive applications is adopted for
geometrical optimization of the tokamak divertor in reactor relevant conditions. The optimization method is implemented in a MATLAB code which solves iteratively the adjoint and forward 2D edge plasma transport equations of the B2 code ("B2_light"). Ultimate goal of this activity is to create a computational divertor design tool applicable (with reasonable computational turnaround times) for DEMO relevant configurations.
Introduction

Conceptual studies on the development of a DEMO fusion reactor have been started in the frame of the European PPPT programme, with the goal to demonstrate significant fusion electricity production in 2050. On the way to DEMO, the ITER tokamak currently under construction is expected to contribute important results in various fields like the demonstration of robust and well-controlled burning plasma regimes, including pulsed operation at power amplification $Q = 10$ and steady-state operation at $Q = 5$, furthermore the test and optimisation of the conventional physics solution for power and Helium exhaust (standard divertor with semi-detached plasma conditions), the validation of test breeding blanket concepts for Tritium production as well as of various technical components and methods, e.g. superconducting magnet technology, remote handling and plasma heating and diagnostic systems.

However, on the way to towards commercial fusion, ITER will leave several important issues either partially or even fully unresolved. Main open issues are first the durability of plasma-facing and structural materials against the challenging heat loads (plasma heating power of DEMO $\sim 4$ times that of ITER), neutron embrittlement and primary erosion (for both the total fluence in a DEMO device is assumed to be $\sim 50$ times that of ITER), second the achievement of Tritium self-sufficiency and finally the demonstration of a substantial net electrical output. Additionally, DEMO will have to improve further significantly as compared to ITER in the fields of stability and control of plasma operation, optimisation of safety issues and reduction of cost. While the well-developed tokamak principle is currently seen as the leading candidate for DEMO, the stellarator concept (W7-X device and HELIAS studies) is pursued in Europe (and outside) as a promising alternative option. After a successful exploitation of W7-X, the stellarator concept is envisioned to become an attractive alternative to the DEMO tokamak due to its intrinsic advantages: steady-state operation, high density operation and current-free plasma configuration without disruptions.

Based on the experience gained in the past decades, the fusion research team at FZJ team is contributing to the development of DEMO within its main areas of expertise in fusion physics and technology.

Within Germany, the three Helmholtz centres IPP (Garching and Greifswald), KIT (Karlsruhe) and FZJ (Jülich) are closely collaborating on conceptual studies towards fusion power production within the German DEMO working group. Since 2010, two meetings of two days duration each were held every year, addressing a variety of topics from both physics and technology.
Additionally, Jülich has been participating in the EFDA Power plant Physics and Technology (PPPT) programme between 2011 and 2013, with contributions to the system studies (SYS) and power exhaust (PEX) tasks.

The main work topics and results are briefly summarized below.

**DEMO system studies**

Within 2013, the software tool for system studies for tokamak fusion reactors, which originally had been developed in the programming language Pascal, has been converted towards a modern Phyton version with graphical use interface. It allows the calculation of the main physics issues of fusion power, heat exhaust and pulse duration, while avoiding time-consuming technological or costing optimisations. The tool has been successfully benchmarked against more sophisticated system codes (PROCESS, HELIOS) and thus allows performing fast and reliable parameter variation studies. Using this software tool, two different DEMO models, a more conservative DEMO model with large aspect ratio $R/a = 5$ and a more advanced DEMO model with $R/a = 3$, each with net electrical output power of $P_{el} = 1$ GW, were developed for further discussion within the German DEMO working group.

**Plasma diagnostic and control**

The reliable operation and control of the plasma in a magnetic fusion reactor requires a robust plasma scenario combined with an integrated diagnostic and control system. Both elements together, scenario and control, have to ensure machine operation in compliance with safety requirements, achieve high plant availability in particular by keeping distance to all known operational limits, and aim for optimized fusion performance while minimizing the aging of components. Initial studies have been performed on the feasibility of DEMO plasma control, including the definition of measurement requirements and identification of candidate diagnostic systems.

Within 2013, a preliminary list of control functions has been developed for DEMO. Essential quantities to be measured and controlled in a tokamak reactor (and mostly also in a stellarator) are the profiles of particle densities, temperatures and plasma current, furthermore the plasma position and shape, plasma radiation, local wall loads and wall temperatures, plasma instabilities, D/T ratio and fusion power. Except for the fusion power, control schemes for most of the other quantities are already available and continuously under improvement on all current major magnetic fusion experiments. However, already for ITER and even more for a future DEMO fusion reactor, the requirements for the reliability of plasma operation are much more demanding than on any existing device. One specific problem is the stationary power exhaust, where the local power flux densities are near to design limits and must be safely controlled to avoid damage to the target plates. Regarding off-normal transient events in a tokamak reactor, the number of high-power disruptions must be minimized towards almost zero, and the few remaining disruptions have to be reliably mitigated, due to the high risk of significant damage to the first wall.
While present magnetic fusion experiments are amply equipped with diagnostic and actuator systems, their implementation on DEMO will only be possible with reduced performance and/or number of systems, due to several reasons: First, the fraction of openings and voids in the breeding blanket has to be minimized in order to achieve a Tritium breeding rate TBR > 1. Second, diagnostic front end components will be subject to a harsh environment (radiation, forces, temperatures etc.) and thus may only be installed at some distance behind the first wall or blanket. Third, available actuators on fusion reactors such as magnetic field coils, auxiliary heating, gas inlets, pellet injectors and pumping systems typically can only provide slow, indirect or weak performance in DEMO. In order to achieve reliable machine operation, enhanced long-term stability of both diagnostic systems and actuators, together with redundancy in terms of both number of methods and number of channels, and finally integrated data analysis together with in-situ calibration and consistency checking methods have to be developed and implemented. Following an initial assessment, microwave diagnostics (ECE and reflectometry), IR polarimetry, neutron and gamma diagnostics are regarded as promising on DEMO, while spectroscopic diagnostics may be feasible only with limited performance due to first mirror lifetime issues.

The feasibility of DEMO diagnostic and control is regarded as particularly difficult when going towards high temporal or spatial resolution, specifically in the divertor region or for core plasma profile measurements. Therefore, the controllability of a more sophisticated (high performance) DEMO plasma scenario will strongly depend on the feasibility of the related diagnostic systems, and substantial R&D on DEMO diagnostic and control has to be launched well in time before freezing DEMO design parameters.

**Disruptions**

Within 2013, the physics properties of disruptions have been analysed and extrapolated towards DEMO conditions. It was found that plasma disruptions on DEMO will release the kinetic plasma energy with a short time of only about 1 – 3 ms (thermal quench), while the inductive energy will be released within a few 10 ms (current quench). While the latter might lead to the generation of a beam of run-away electrons which may eventually hit the first wall and cause local damage (melting), the thermal quench might also approach melting conditions in case of a non-uniform radiation distribution. Therefore it has been concluded that a reliable system for avoidance and mitigation of disruptions has to be developed for DEMO (and to a large extent already for ITER high performance discharges).
Journals refereed

1. Abdullaev S.
   **On collisional diffusion in a stochastic magnetic field.**

2. Behr W, Faidel D, Fischer K, Pap M, and Offermanns G.
   **Welding feasibility study of U-shape lips at ITER Port-Plug with new laser beam sources.**

   **ICRF specific plasma wall interactions in JET with the ITER-like wall.**

   **Residual carbon content in the initial ITER-Like Wall experiments at JET.**

   **Fuel retention studies with the ITER-Like Wall in JET.**

   **Fast shutter concepts for the new ITER core CXRS upper port plug baseline considering the actuator located inside and outside the port plug.**

   **A wide angle view imaging diagnostic with all reflective, in-vessel optics at JET.**

   **Longterm Evolution of the Impurity Composition and Transient Impurity Events with the ITER-like Wall at JET.**

   **Evolution of surface melt damage, its influence on plasma performance and prospects of recovery.**


**Determination of structure tilting in magnetized plasmas—Time delay estimation in two dimensions.**  

**Global migration of impurities in tokamaks.**  

**Impact of the ITER-like wall on divertor detachment and on the density limit in the JET tokamak.**  

**A new radiation-hard endoscope for divertor spectroscopy on JET.**  

**Investigation of the Impact on Tungsten of Transient Heat Loads Induced by Laser Irradiation, Electron Beams and Plasma Guns.**  

**Investigation of the Impact on Tungsten of Transient Heat Loads Induced by Laser Irradiation, Electron Beams and Plasma Guns.**  

**Simulations of tungsten transport in the edge of J ET ELMy H-mode plasmas.**  

**Kinetic effects of inclined magnetic field on physical sputtering by impurity ions.**  


Advanced materials characterization and modeling using synchrotron, neutron, TEM, and novel micro-mechanical techniques—A European effort to accelerate fusion materials development.

First studies of ITER-diagnostic mirrors in a tokamak with an all-metal interior: results of the first mirror test in ASDEX Upgrade.

Dust investigations in TEXTOR: Impact of dust on plasma-wall interactions and on plasma performance.

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40. Mekkaoui A.

41. Mekkaoui A.

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Using the radiation of hydrogen atoms and molecules to determine electron density and temperature in the linear plasma device PSI-2.

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Rotation dependent ion fluxes in front of resonant magnetic perturbation coils.

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63. Tokar M, and Koltunov M.
A simplified, numerically verified model for the global plasma reaction on a local cooling.
64. Tokar MZ, and Koltunov M.  
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67. Wirtz M, Linke J, Pintsuk G, Singheiser L, and Zlobinski M.  
**Comparison of Thermal Shock Damages Induced by Different simulation Methods on Tungsten.**  

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**Experimental Observation of a Magnetic-Turbulence Threshold for Runaway-Electron Generation in the TEXTOR Tokamak.**  

**Hydrogen retention in tungsten materials studied by Laser Induced Desorption.**  

**Journals (not refereed)**

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30. Salmagne C, Reiter D, and Gibbon P

**Grid-free Tree-code Simulations of the Plasma-Material interaction region.**

31. Schlummer T, Marchuk O, Bertschinger G, Biel W, and Reiter D

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34. Steinbusch B, Gibbon P, Reiter D, and Sydora R
Gyro-resolved nonlinear kinetic effects at topologically complex plasma-material interfaces.

35. Wegener T, Coenen J W, Philipps V, and Unterberg B
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Role of ion-atom collisions in the fast-beam diagnostics of fusion plasmas.

Fuel retention and long term outgassing with the ITER-like Wall.

Quantitative modeling of fuel retention in the JET-C and JET-ILW wall configurations by WallDyn and predictions for ITER.

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