9th International workshop on: Stochasticity in Fusion Plasmas (SFP) titled 3D physics in stellarators and tokamaks

3D physics in stellarators and tokamaks

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Seminar information

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Over the last six decades, research into controlled nuclear fusion through the magnetic confinement of hot plasmas has been pursued in many countries with the long-term aim of providing a practically inexhaustible, safe and environmentally friendly energy source

To date, the most advanced concepts for fusion confinement are the tokamak and the stellarator. The latter is exemplified by the Wendelstein 7-X (W7-X) stellarator in Greifswald, Germany, which has recently completed its operational phase 1.1. The tokamak scheme is currently the leading variant for plasma confinement and is therefore being used for the next-generation device ITER and is planned for the DEMO reactor. However, reliably achieving and maintaining the stability of fusion plasmas remains an important fundamental issue that needs to be resolved.

In both tokamaks and stellarators, stochastic magnetic fields can arise and influence the interplay between three-dimensional (3D) magnetic topology and plasma confinement. Stellarator devices represent an inherent three-dimensional challenge. They make use of the island divertor concept, and stochasticity and magnetic topology therefore play a fundamental role in their operation. With the extended operational regimes pioneered on the Large Helical Device (LHD) and with W7-X, attention has been directed towards the challenge of 3D plasma equilibria, transport and plasma–surface interactions.

In the tokamak line, non-axisymmetric magnetic perturbations, which change the magnetic topology, are applied on the majority of large-scale tokamaks nowadays to control plasma edge stability and transport. Recent research has highlighted the significance of the role that stochasticity and 3D magnetic topology also play in this fundamentally 2D concept. Their influence can be seen in transport and energy confinement, in the nature of disruption events and in the control of various magnetohydrodynamic (MHD) instabilities, most notably edge-localized modes (ELMs), which expel considerable amounts of energy from the plasma and pose a risk of damaging plasma–facing components in ITER and other next-generation fusion devices.

The existence of these stochastic and 3D effects brings tokamak and stellarator physics closer together, and a holistic approach to studying them provides the most promising path to making good progress. Understanding these effects is essential for the success of future fusion devices, and they represent a hot topic in current fusion research. In addition, reversed field pinches offer access to these topics with unique features such as the bifurcation into self-generated 3D equilibria and multi-mode unstable plasma conditions with a high degree of magnetic field stochasticity. Joint discussions of these aspects across the three communities will foster progress on basic as well as applied understanding in these complex branches of high-temperature plasma physics. Therefore, it will be of great interest and scientific importance to share the most up-to-date theories and techniques and to provide a platform for discussion between leading experts in the field.

The 9th International workshop on "Stochasticity in Fusion Plasmas (SFP)" is an attempt to discuss issues relating to impact of 3D magnetic fields on hot plasmas from all sides, bringing together experts from different devices (tokamaks, stellarators and reversed-field pinches) and from different fields (equilibrium and confinement, turbulence, MHD instabilities, transport and plasma–wall interactions). The previous seminars focused on integrating experimental methods of controlling MHD instabilities, on comparing the relative merits of 2D and 3D conceptualizations of hot plasmas, and on the underlying physics of stochastic effects. In contrast, the focus of this seminar will be on the latest experimental results from stellarators and tokamaks. This will naturally lead to a comparison of W7-X's 3D island divertor with various different axisymmetric divertor configurations under the application 3D fields being a highlight of the seminar.

The success of these four previous Workshops is exemplified by the publication of associated special issues of the journal Nuclear Fusion. The 597th WEH Seminar absorbed the International Workshop on Stochasticity in Fusion Plasmas, which had run successfully for over a decade and had established itself as the foremost international workshop in the field, the plan is now to continue this history independently of the WEH foundation.

For the most recent seminar successful publications were made in a special issue of the IOP journal "Plasma Science and Technology", which gave especially phd students a venue to publish their research in a peer reviewed journal.

Discussing together and summarizing recent approaches will improve the physics understanding of various effects of 3D fields in magnetically confined plasmas. Analyzing the influence of stochasticity and magnetic topology in fusion plasmas will be beneficial for research in the field and will guide the design of future fusion devices. A further major goal of this seminar is to give young scientists the opportunity to enter an active and growing field of research by interacting with world-leading experts.

Contents

Invited Talks

Results from the first operation phases of Wendelstein 7-X

Thomas Sunn Pedersen, for the W7-X Team.
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Stellarators provide a potentially attractive concept for fusion power production, owing to their intrinsic steady-state capabilities, and their lack of runaway electron issues. However, high confinement at high ion temperatures has in the past been an elusive goal, primarily owing to prompt orbit losses. Wendelstein 7-X (W7-X) is a highly optimized stellarator experiment that went into operation in 2015.
Its optimization goals include not only a strong reduction of the just mentioned prompt orbit losses, but also stable MHD equilibria at volume-averaged beta values up to 5%, a reduced bootstrap current, and successful operation of the island divertor at high performance. With a 30 cubic meter volume, a superconducting coil system operating at $2.5\,\mathrm{T}$, and steady-state heating capability of eventually up to $10\,\mathrm{MW}$, it was built to demonstrate the benefits of optimized stellarators at parameters approaching those of a fusion power plant.

Operation phase 1.2 featured the full complement of 10 divertor units, ECRH heating with up to 10 gyrotrons, more than 30 diagnostic systems, and a pellet fueling system. This talk will give a general overview of the W7-X goals and capabilities, and describe results from divertor operation, including measurements and corrections of error fields, characterization of divertor heat loads, evidence of detachment, and performance extension into regimes with high densities ($n_\mathrm{e} \approx 10^{20}\mathrm{meter}^{-3}$) and high ion temperatures ($T_\mathrm{i} \approx 3.5\mathrm{keV}$), long pulse lengths (up to $100\,\mathrm{s}$), and high stored energies ($E > 1\,\mathrm{MJ}$). Tests of the W7-X optimization will also be reported.

Results from the first operation phases of Wendelstein 7-X

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During the first divertor operation phase of the stellarator Wendelstein 7-X (W7-X), a 2D radiation distribution in the triangular plasma cross-section has been measured both before and after wall boronisation. A clear up-down asymmetry in radiation intensity has been frequently observed in spite of a symmetric magnetic topology.

The position of the maximum intensity varies, both radially and poloidally, depending on the plasma parameters and the magnetic configurations that respond to the 3D SOL structure. With increased plasma density, the radiation zone shifts from the SOL toward the last closed flux surface, accompanied by an increase in total radiated power. Under certain conditions, stable, strongly radiating regimes exist, characterized by a high radiation loss fraction, $f_{\mathrm{rad}} > 90\%$, a strong reduction of the heat load on the divertor targets (i.e. power detachment) and even more by a favorable, lower plasma confinement degradation. Further increases in density could lead to uncontrollable radiation collapses with precursors, such as pronounced poloidal asymmetric radiation patterns and simultaneous deterioration of plasma confinement. In addition, this asymmetric radiation pattern and its location also change with the dominant impurity species (responsible for the main power loss) and their locations. Experimental observations by a two-camera bolometer system in combination with spectroscopic diagnostics at W7-X are described.

The analysis of this phenomenon currently focuses on ECR-heated hydrogen plasmas, which contain the following radiators: 1) intrinsic impurities (such as carbon and oxygen) released from the wall and targets by wall outgassing and sputtering, 2) extrinsically seeded gases (such as N_2, Ne and Ar) for special experimental purposes, and 3) metallic elements (such as Fe and Cu) injected by LBO and TESPEL for transport studies. The physics behind these observations is presently analyzed using the impurity transport codes STRAHL and EMC3-EIRENE and the results will be presented.

Plasma-Wall Interaction in W7-X with uncooled graphite divertor

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The stellarator Wendelstein 7-X completed its initial operation with ten divertor modules made of uncooled graphite plasma-facing components PFCs The plasma exhaust concept and first wall properties are substantially different from those in the initial campaign which had limiters and explicitly avoided edge magnetic islands. The ten divertors now intercept an edge island chain. Additionally the inner vessel has been complemented by adding graphite PFCs to the already installed CuCrZr heat shields. Before the initial divertor operation (OP1.2a) started, the device vessel was baked up to 150° °C, and glow discharges in helium and hydrogen were performed to condition the device. ECRH discharges were conducted in helium in order to reduce the impurity and hydrogen content to acceptable values for steady and long pulse operation. The main impurities have been identified spectroscopically to be carbon and oxygen in the plasma-edge layer whereas the post-plasma outgassing also includes methane, water and carbon monoxide. Water was attributed to residual water in the graphite PFCs and was released at higher surface temperatures. Indeed the release of O and subsequent impurity sputtering of O on C determined the high radiated fraction in the edge plasma and limited the operational window. Application of boronisation cured the challenging conditions by backing the O cycle and permit an expansion of the operational window. Subsequently, higher divertor density operation, low radiated fraction operation and long-pulse operation was achievable.

Key studies in the full 3D geometry are related to the determination of the impurity production distribution with respect to spatial distribution and absolute strength. In parallel to spectroscopic observations in the edge layer in different plasma configurations also dedicated experiments related to the identification of the material migration footprint were conducted. Most relevant with this respect was a global tracer injection in the standard iota= $5/5$ configuration under attached divertor conditions. 13CH4 methane was injected in 30 identical discharges of about $10\,\mathrm{s}$ each through a divertor gas injection system simulating a divertor C source. Optical emission spectroscopy and post-mortem analysis of divertor modules are going on the identify locations of erosion and deposition as well as to identify the amount of C which reaches the plasma core, thus, is unscreened from the divertor. Secondly, marker layers were implemented at different locations in the divertor permitting access to local erosion and deposition at multiple positions in one divertor as well as at different divertors. Finally, impurity injection experiments in different configurations were applied to determine the source strength and relate it to the global content of C (and O).

These dedicated studies in the second operational campaign are related to the first campaign and post-mortem analysis by different techniques providing information about erosion/deposition/migration as well as fuel content and layer formation. Revealing overall a very strong erosion at the main interaction zones at the strike-line locations and no clear location of massive re-deposition. Likely baffle areas are turned into main deposition areas as laser-based diagnostics and colorimetry of wider areas suggested.

Loads on plasma wall components in W7-X and corresponding limits

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The third experimental campaign of the stellarator experiment Wendelstein 7-X has been completed 2018, October 18th. During the experimental programs, starting with the first operation phase OP1.1 (Dec. 2015 – March 2016 [1]) up to OP1.2a (Sept. 2017 – Dec. 2017 [2]) and OP1.2b (July 2018 – Oct. 2018 [3]), the energy input and plasma performance as well as the heat and particle loads onto the in-vessel components have been increased continuously. In OP1.1 five limiters on the inner wall were used, in OP1.2 ten inertially cooled divertor units (with fine grain graphite as plasma facing surface) have been installed in the plasma vessel. The heat loads and the resulting temperature rise have been closely supervised by thermography systems but also by using a large number of thermo-couples incorporated in many in-vessel components. A fine adjustment of the currents in the main superconducting magnet system together with optimized current settings of the control and trim coils were required in order to establish a certain magnetic field configuration which enabled a symmetric load on the divertor units, avoiding at the same time excessive loads on other wall components. Finally, an energy input of about 200 MJ could be achieved obeying the pre-defined load and temperature limits for the different plasma facing components. A survey of the visual inspection results after OP1.2b will be given in context of the performed plasma experiments (the figure shows the view onto one of the ten divertor units with the complex pattern of erosion and deposition zones).

Dedicated experiments aiming at controlled overheating of selected target modules have been performed to investigate the plasma and divertor performance under the condition of strong impurity emission. Such emission of carbon atoms and of clusters/particles from small spots at leading edges of single target fingers have been observed by a fast video camera system equipped with a CIII filter. The effect on the central plasma parameters as well as on the core radiation was rather negligible, probably, due to the good impurity retention in the divertor plasma. The surface temperatures measured by the IR cameras turns out to be rather low (max. 2200 °C), but could be undervalued because it could be out of the linear range of the sensor and/or due to the limited spatial resolution of the IR cameras. First estimations regarding the spot size of the overheated regions and the amount of emitted particles will be

presented.

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ELM control by 3D magnetic perturbations

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It is necessary to avoid large transient local heat load on the divertor induced by Edge Localized Mode (ELM) to achieve long pulse steady state operation in a tokamak fusion reactor. Uncontrolled type-I ELM is expected to be a significant issue to damage the materials of plasma facing components in ELMy H mode plasmas in future fusion device like ITER. Based on present materials considered as divertor plate and first wall, the crash amplitude of type-I ELM in ITER baseline operation is expected to be reduced at least by a factor of 20.

Various methods have been developed in the past for ELM control in tokamaks. Resonant Magnetic Perturbation (RMP) is one effective method for ELM control in present tokamaks. It is usually generated by external non-axisymmetric magnetic field coils. Significant efforts have been made in recent years and the control effects have been successfully observed in many tokamaks. Edge ergodization due to the overlapping of the neighboring islands induced by RMP has been at first proposed as an explanation of the ELM suppression. However, plasma response may shield the resonant components and may not form the edge stochastic layers. Significant progresses, especially on plasma response, have been made in the understanding of the mechanism of ELM control using RMP in the recent years. These enhanced the confidences in the application of RMP for ELM control in ITER in the future. To apply RMP for ELM control in future fusion device, like ITER, there are still a lot of challenges. For example, at first, it is necessary to extend ELM suppression in low torque, low collisionality plasmas.

Presently, most of the ELM control experiments were done in high torque with dominant NBI heating. It is still a challenge to access ELM suppression in low torque discharge, in which field penetration of the core resonant harmonics may limit the plasma performance. Secondly, how to optimize the coil configuration for accessing ELM suppression and avoiding performance degradation? Plasma response plays a key role in accessing ELM suppression. It strongly depends on plasma operational regime and coil configuration. The physical understanding the difference between

ELM mitigation and suppression is also a critical issue. Thirdly, steady heat flux control during ELM suppression is also necessary. The 3D edge magnetic topology may enhance local heat flux on the divertor that connected to the inner flux surface with high temperature.

Finally, it is necessary to extend the ELM suppression in long pulse operation with mental wall. It needs to prove that ELM suppression and heat flux control by RMP can be sustained in long pulse. The heavy impurity control is also an important issue for long time operation with ELM suppressed. The progresses attempted to solve these challenging issues together with the progresses in ELM control in the EAST tokamak are overviewed.

Magnetic reconnection in 3D equilibria of RFX-mod

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The reversed field pinch (RFP) device RFX-mod features strong internal transport barriers when the plasma enters states with a single dominant helicity. Such transport barriers enclose a hot helical region with high confinement whose amplitude may vary from a tiny one to an amplitude encompassing an appreciable fraction of the available volume. In this work it is shown that the position and the width of the stochastic boundary encompassing the thermal structures can be estimated using the concept of 3D quasi-separatrix layer (QSL) [1]. Such concept has been developed in solar physics to explain the occurrence of reconnection phenomena in solar flares without true separatrices.

The QSL concept is novel to laboratory plasmas and has been adapted to their toroidal geometry. The application of the QSL model indicates that the transition from narrow to wide thermal structures is mainly due to the progressive stabilization of secondary modes, although a role is also played by the progressive strengthening of the equilibrium that occurs as long as the dominant mode increases.

In particular, these results open promising scenarios for RFX-mod2 [2], the upgrade of the RFXmod device presently underway, where, due to a new magnetic front-end, reduced secondary modes are expected, which will produce larger thermal structures. Furthermore, this first application of the quasi-separatrix layer to a toroidal plasma indicates that such a concept is ubiquitous in magnetic reconnection, independent of the system geometry under investigation.

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Edge diagnostic development for Wendelstein 7-X

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In 2015, the magnet system of Wendelstein 7-X was successfully commissioned proving reliable operation of the superconducting bus-bar system that is the major contribution by Jülich to the machine construction. Based on the excellent technology expertise developed at TEXTOR, JET and many other devices, Jülich is also leading the research field of PWI at W7-X covering both edge plasma and surface characterization. For this purpose, a set of diagnostics is developed. In particular, the multi-purpose manipulator for probes with slow and fast drive, cooling and gas feed; microwave correlation reflectometry; versatile gas inlet system in divertor region; core plasma X-ray and VUV spectroscopy were successfully commissioned in the very first plasma campaign already allowing for physics studies in limiter operation. Diagnostic development is continued with versatile endoscopes for local highly resolved divertor observation with the option of 2D tomographic reconstruction, several probe heads for probing edge plasma, feeding gas locally, and exposition of material samples including corresponding probe electronics for testing divertor operation of W7-X. A concept of a divertor manipulator for probing and long pulse exposition under divertor conditions is prepared for operation with water-cooled divertor in the next campaign. This will include a versatile observation system and laser for in-situ surface analysis. A number of surface analysis techniques are also available for post mortem analysis of wall elements. By this, W7-X will be equipped for pioneering PWI research allowing also preparation of operation with a Tungsten wall.

Physics Design of Stellarators

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New frontiers in stellarator optimization allow for design of next generation experiments and reactors with capabilities exceeding current devices. In this talk we discuss the stellarator design optimization procedure and recent improvements in metrics for 1) Energetic particle transport, 2) Turbulent transport, 3) Coil design and 4) Divertor design for stellarator reactors. Optimization codes modify the coefficients of the plasma boundary, solve for the plasma equilibria, and then apply penalty functions based on desired plasma properties. Some examples of penalty functions are levels of neo-classical transport, metrics of quasisymmetry, etc. We will briefly discuss some of the historical metrics used to design the optimized stellarators of W7-X, HSX and NCSX. We also point out some of the pitfalls that exist in the current optimization schemes, including local minima and non-convex spaces Next generation stellarator experiments will require demonstrating significant improvement in various topical areas. We show that using energetic particle metrics developed by V.V. Nemov, we can optimize quasi-helically symmetric equilibria to have excellent energetic particle confinement. New frontiers in turbulent optimization produce metrics that target turbulent heat flux by increasing the coupling between unstable and stable modes. We show that it is now possible to design configurations optimized for low ITG turbulence. Significant development has been made in new algorithms for coils. We look at some of these methods and their capabilities. We also show the connections between coil design, energetic particle transport and divertor design. For divertors, feasible solutions are strongly tied to configuration type. We will discuss several 3D divertor types in current machines, and ones that are thought to be viable for future devices.

3D modelling of dissipative divertor conditions

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The compatibility of detached divertor plasmas with application of resonant magnetic perturbations (RMPs) for control of edge localized modes (ELMs) remains a key challenge for magnetic confinement fusion. Non-axisymmetric particle and heat loads onto the divertor targets appear as a result of RMP application, and thus require a three-dimensional (3D) plasma edge model. However, no stable 3D numerical model was available for assessment of detached divertor conditions with RMP fields so far. Recently, the numerical framework within EMC3-EIRENE has been improved which now enables numerical access to detached divertor plasmas for the first time. Numerical stabilization is achieved by a linearization of the energy losses from interaction with neutral gas (i.e. from ionization and molecular dissociation). These energy losses are very sensitive to the electron temperature in this regime, and the improved numerical scheme includes a partially implicit method for a more accurate treatment. To validate the model for detached divertor plasmas, we present a comparison between EMC3-EIRENE and SOLPS-ITER for an unperturbed configuration targeting the initial non-active phase in ITER with an L-mode, gas fueled $1.8\,\mathrm{T}/5\,\mathrm{MA}$ hydrogen plasma at $q_{95} = 3$, and demonstrate that both codes show a similar particle flux roll-over during a gas puff (density) scan.

For ITER, EMC3-EIRENE simulations for a low power ($30\,\mathrm{MW}$) H-mode discharge anticipated for the initial non-active operation phase show a decoupling of divertor loads around the original strike zone (OSZ) from those further away in the non-axisymmetric strike locations: an earlier transition to detachment is found at the OSZ while the latter locations remain attached. While detachment is commonly characterized by relating divertor conditions to upstream conditions (e.g. at the outboard midplane position of the separatrix), such a characterization is severly complicated by RMP application. This is because field lines can connect the edge of the nominal confined region to the divertor targets resulting in non-axisymmetric upstream and downstream conditions. The single fluid resistive MHD model MARS-F is used in this analysis to include plasma response effects in the EMC3-EIRENE simulations. The calculated plasma response leads to screening of most of the resonances, and consequently a narrower region of broken perturbed flux

surfaces with connection to the divertor targets with respect to the vacuum RMP approximation. However, resonant field amplification near the separatrix results in non-axisymmetric footprints which can be of similar size or even larger than those found in the vacuum RMP approximation depending on the level of toroidal plasma rotation.

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Overview of 3D effects on the plasma confinement and transport in magnetically confined fusion

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3D effects on the plasma confinement and transport in magnetically confined fusion plasma both in helical and tokamak configuration are discussed in this talk. There are two types of 3D effects in the magnetically confined fusion plasmas. One is the 3D magnetic flux surface effect in helical plasmas. Here, the 3D means a lack of toroidal symmetry of magnetic flux surface and only the transport perpendicular to the magnetic flux surface plays a role as well as tokamak, where the magnetic flux surface has a toroidal symmetry. The 3D magnetic flux surface effect is crucial in helical plasma but not in tokamak except for the plasma with resonant magnetic field perturbation (RMP). The other is the 3D topology effect in the stochastic region, magnetic island, and the plasma boundary. Here, the 3D means a lack of nested magnetic flux surface and the mixture transport parallel and perpendicular to the magnetic field become important. The 3D topology effects are common issue both in tokamak and helical plasma.

Recently the role of topology 3D effect is recognized to be essential. For example, the transport in the radial direction in the region with a stochastic magnetic field is determined by the combination of parallel and perpendicular transport. The power fall-off length in the Scrape-off-layer (SOL) is determined by the balance between the two transports parallel and perpendicular to the magnetic field. In the magnetic island, the magnetic field becomes partially stochastic near the X-point, while it has nested magnetic flux near the O-point of the magnetic island. Therefore, 3D topology effects always exist in the plasma with spontaneous magnetic island and induced magnetic island by resonant magnetic perturbation (RMP).

The following experimental results are reviewed. (1) The bifurcation of the transport and turbulence spreading in the magnetic island in tokamak plasma [1,2]. (2) The electron and ion heat and momentum transport in the stochastic region in helical plasma [3,4]. (3) The helium transport and turbulence exhaust at the boundary between the closed and open magnetic field i.e. transport across the last closed flux surface (LCFS) [6]. Turbulence spreading into the SOL region and the impact on the power fall-off length is also discussed as an example of topology 3D effect on transport [7,8].

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Seeding of tearing modes by external and internal 3D perturbations

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In tokamak plasmas, large internal events (sawteeh, fishbones, etc) are able to provide the drive for magnetic reconnection at the neighbouring resonant surface, which lead to seeding of tearing modes. This type of tearing mode formation is considered to be one of the most dangerous for future fusion reactors like ITER. Recent findings show that the crash leads to the formation of an ideal kink mode with large saturated amplitude at the resonant surface immediately after the sawtooth or fishbone [1,2]. This kink mode transforms into a tearing mode on a much longer timescale than the crash itself. Thus, the ideal kink mode, formed at the resonant surface after the crash, provides the driving force for the magnetic reconnection. These experimental findings from ASDEX Upgrade and DIII-D tokamaks were also confirmed by non-linear two fluid MHD code [3].

The process of the island formation due to an internal event is similar to the seeding of the tearing modes by external perturbations in the presence of differential plasma rotation [4]. In this case, the perturbation modify the plasma rotation in a way that the differential rotation between electron fluid velocity at the resonant surface and the perturbation is reduced. When this differential rotation vanishes, the electrons are at rest with respect to the external perturbation and magnetic flux efficiently penetrates into the plasma. This flux penetration is accompanied by fast island growth and finally, the non-linear evolution leads to saturation of the island size. These results show importance of the differential plasma rotation both from internal and external seeding of tearing modes.

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RMP ELM suppression impact on divertor heat and particle fluxes at ITER-like conditions

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RMP ELM suppression experiments at ITER-like conditions (shape, collisionality, RMP spectrum) in DIII-D show little splitting of the heat flux to the divertor targets, despite robust splitting in the particle flux. This lack of divertor heat flux splitting is a potentially important result for ITER because splitting of the divertor heat flux into multiple lobes displaced from the primary strike point could complicate heat flux handling during RMP ELM suppression in ITER and other tokamaks with tight divertor baffling. In DIII-D, strike point splitting is routinely observed in the divertor particle flux during RMP operation. The observed splitting is consistent with the toroidal mode number n of the perturbation, but the measured separation of the divertor particle flux lobes exceeds predictions of a vacuum (TRIP3D, MAFOT) or ideal plasma response (VMEC-XPAND) model by factors of 3-5. The large particle flux lobe separations present a challenge for plasma response modeling, because the predicted response using linear, resistive MHD simulations is dominantly a screening response, which should reduce the divertor lobe splitting below the vacuum model predictions. However, nonlinear resistive MHD simulations (JOREK) of these discharges predict larger particle flux lobe separation and show good comparison with experimentally measured splitting.

Current ramps, which were limited in amplitude for a subset of RMP coils to be consistent with force limits on the RMP coils in ITER, were used to modify the divertor lobes. The particle flux lobes changed during the RMP current ramps, but the heat flux profile was not affected, consistent with the lack of heat flux lobe structure. The lack of clear lobe structures in heat flux is now understood to be due to an increase in the volumetric C radiation in the inner divertor as the peak heat flux drops to $< 2\,\mathrm{MW/m^2}$. Possible synergistic effects of impurity gas injection and RMP current ramps were also examined using neon and argon gas injection into the ELM suppressed phase. Both gases produced stable radiating mantles between $0.95 \leqslant \Psi \leqslant 1$, a 60% radiated power fraction, and significantly reduced heat flux

to both strike points while ELM suppression was maintained. These results show that RMP ELM suppression in ITER-like conditions is compatible with an impurity radiation-enhanced boundary.

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3-D characteristic of disruptions in tokamaks

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Disruptions, which is a rapid loss of the plasma energy and current, can impose significant thermal and electromagnetic loads on the tokamak structure. Runaway electrons are expected to be generated due to the rapid loss of the plasma current and an uncontrolled loss of highly energetic electron beam to the plasma facing components might lead to serious damages. These consequences of disruptions require careful design of tokamak components to ensure that they reach the projected lifetime. Disruptions are intrinsically of 3D characteristic because the instabilities causing disruptions are non-symmetric. The consequences of disruptions, including heat loads, electromagnetic loads and runaway electrons, can be 3D. For heat loads and runaway electrons, the localization leads to higher local heating. Electromagnetic loads can impose global tilting forces on the vacuum vessel and also localized forces which might damage the components within the vessel.

A disruption mitigation system has presently been invoked to mitigate the damaging effects of disruptions. Massive amounts of noble gases are rapidly injected to the plasma, which radiates the plasma energy and spreads it more uniformly, therefore lowering the peak heat loads. A rapid loss of the plasma current caused by the injected gas lowers the electromagnetic loads, but this might also increase risk of runaway electron generation. Since the gas is injected from a limited number of locations there are still issues of 3D localization of heat loads, which has to be considered in the design of the disruption mitigation system.

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Interaction between runaway electrons and resonant magnetic perturbation during disruptions on J-TEXT tokamak

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High-energy runaway electrons generated during the plasma disruption could result serious damage to plasma-facing components. The next generation fusion machines, like ITER and DEMO, will need a reliable method for controlling or suppressing runaway electrons. Previous experimental results show that the massive gas injection can't provide enough impurities to achieve robust runaway suppression due to low gas mixture efficiency and extreme high Rosenbluth density for runaway avoidance.The transport of runaway electrons is affected by the magnetic perturbation. Robust runaway suppression has been reached on J-TEXT with mode penetration or mode locking by the application of resonant magnetic perturbation (RMP) with m/n=2/1 before the thermal quench. The strong stochastics in the whole plasma cross section expel out the runaway seed and results in runaway free disruptions on J-TEXT. This provides alternative runaway suppression during disruptions for large scale tokamak. It is found that hydrogen supersonic molecular beam injection has the capacity to eliminate RE current by provoking magnetic perturbations which increase RE losses rapidly.

3D Linear Stability Studies for Tokamaks and Stellarators

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Three-dimensional stability studies for stellarator and tokamak equilibria play an important role for the analysis and interpretation of experimental results, and the design of new fusion devices. The axisymmetry of tokamak configurations may, for example, be broken, by the fields of additional magnetic perturbation coils which are applied to mitigate or even to suppress Edge Localized Modes (ELMs).

The linear stability CASTOR3D code [1] is a very flexible and versatilely applicable numerical tool to investigate 3D equilibria. The code solves an extended eigenvalue problem composed of the linearized MHD equations, and equations describing the vacuum and external conducting structures therein, e.g. ideal and/or resistive walls. The latter equations are derived from a vacuum energy functional of the surface (plasma surface, wall structures) and coil currents using the thin wall approximation [2]. Depending on their complexity, the conducting structures are either discretized by a spectral method, or by triangular finite elements where the resistivity in each triangle may be different. The linearized MHD equations describe several physical effects, such as, plasma resistivity, plasma flow, and parallel viscosity.

In quasi-axisymmetric configurations the rotational transform is partly produced by coil currents, while the magnetic field strength is approximately independent from the toroidal Boozer coordinate. That is, these configurations have common properties with tokamaks and stellarators. We investigate the effects of plasma flow (in direction of the quasi-symmetry), parallel viscosity, and resistive wall structures on the growth rates of ideal external kink modes for such a quasi-axisymmetric configuration, and compare the results with those obtained for an axisymmetric approximation of this equilibrium. We will show that in the 3D case the modes are locked for small rotational frequencies, and start to oscillate for frequencies larger than individual threshold frequencies of the modes. Furthermore, coupled tearing modes are studied for a 3D ASDEX Upgrade-type configuration including effects of toroidal flow and viscosity, and Wendelstein 7-X-type equilibria. There is experimental evidence that instabilities occur in Electron-Cyclotron Current Drive (ECCD) discharges in W7-X [3].

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Linear and non-linear analytical and numerical calculations of edge harmonic oscillations in ELM-free H mode plasmas

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An area of crucial importance for successful plasma scenarios in tokamaks is the development of ELM-free H-modes. A future reactor will probably be intolerable to regular ELMs, even small deliberately paced ELMs. The high performance QH-mode regime [1] is a particular advance because it is ELM free. The bootstrap current at the pressure pedestal forms an extended region of low magnetic shear, which causes a susceptibility to non-resonant ideal fluctuations due to the weakened restoring force associated with magnetic field line bending. These so called edge harmonic oscillations (EHOs) saturate in amplitude, creating therefore new equilibria in the reference frame of the slowly rotating mode. It has recently been shown [2] that the perturbations are non-resonant pressure driven infernal modes, where the plasma displacement of the upper sideband of the nearly resonant mode forms the observed plasma edge corrugation, and the magnetic field associated with the upper sideband extends into the vacuum. The so called external-infernal mode vanishes if a perfectly conducting wall is placed on the plasma edge. The linear analytic model of external infernal modes can also be extended to show that while EHOs can be excited, high n modes associated with ELMs are stabilised by consideration of plasma flows and finite Larmor radius effects [3]. Nevertheless, the non-linear saturated amplitude of the EHOs is clearly an important feature that cannot be established with a linear model. It has recently been shown [4] that the 3D free boundary code VMEC recovers the saturated amplitude of external kink modes with traditional monotonic q-profiles [5]. Further advances [6] show that these current driven external kink modes can be eliminated with a suitable choice of the q-profile, and that the VMEC free boundary code can establish [6, 7] the non-linear extension of the external-infernal modes identified in Refs. [2]. The conditions (pressure gradient and q-profile) for obtaining the EHOs, and the detailed mode structure, are shown to agree across linear analytic theory, the KINX ideal MHD code, and the free boundary 3D VMEC code. The work presented here will show how analytic theory can isolate the physical mechanisms for excitation of EHOs, can guide efficient multi-dimensional numerical simulations, and can hopefully help experimentalists establish robust ELM-free H modes in ITER and DEMO.

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Physics studies with the ICRH system for the Wendelstein 7-X stellarator

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The operational phase 2 (OP2) of the superconducting stellarator Wendelstein 7-X (W7-X) at the Max-Planck-Institute in Greifswald is being prepared, allowing plasma pulses of up to 30 minutes with max. $10\,\mathrm{MW}$ $140\,\mathrm{GHz}$ ECRH as main heating system. An important aim of this optimised stellarator is to demonstrate fast ion confinement at volume averaged beta values up to 5% [1], corresponding to plasma densities above $10^{20}\,\mathrm{m}^{-3}$. To study the confinement of alpha particles under those conditions in a future helias reactor, energetic ions with energies in the range 100 keV are required in the core of W7-X [2]. Such a population can be created using Ion Cyclotron Resonance Heating (ICRH) using various heating schemes, including the newly demonstrated 3-ion heating scenario [3, 4].

The ICRH system under construction for W7-X aims at delivering up to $1.5\,\mathrm{MW}$ RF power at $25-38\,\mathrm{MHz}$ in pulses of up to $10\,\mathrm{s}$ [5]. The shape of this 3D-shaped antenna is carefully matched to the 3D shape of the Last Closed Magnetic Surface (LCMS) of the standard magnetic field configuration on W7-X [6], resulting in a variable curvature in toroidal and poloidal direction. Two RF generators will feed the antenna, allowing for a large flexibility in strap phasings, e.g. $(0, \frac{\pi}{2})$, and increased power coupled to the plasma. Plasma wall conditioning studies are included in the future physics programme. The paper will describe the antenna set-up, the possibilities it offers for heating, fast particle generation and wall conditioning stud-

ies, the need for fast particle and edge diagnostics, and the various options it offers for fusion material studies under real stellarator plasma conditions.

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Contributed submissions

The edge turbulence transport during the ELM mitigation and suppression with n=1 resonant magnetic perturbation on EAST

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The edge localized mode (ELM) has been mitigated and suppressed by the n=1 resonant magnetic perturbation in EAST tokamak. The edge SOL turbulence structure is measured by the reciprocating probes, and the radial turbulent transport is calculated by the four-tip Langmuir probes. During the ELM mitigation and suppression, the background turbulence is enhanced significantly and leads to a large amount of outward turbulent particle flux. Simultaneously, the ion saturation current at divertor plates increases obviously during the ELM control phase, which is consistent with the enhanced edge particle transport in the outer midplane. In consequence, a new channel for particle exhaust primarily induced by the broadband turbulence during the ELM mitigation and suppression is formed.

Spectroscopic characterization of boronizations in Wendelstein 7-X

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The second experimental campaign in divertor configuration employing uncooled graphite plasma-facing components had been successfully conducted at Wendelstein 7-X. A key achievement was the plasma operation at high core densities exceeding $10^{20}\,\mathrm{m}^{-3}$ due to the reduced radiation-induced density limit. Such a reduction was achieved due to the application of boronizations – the in-situ plasma-chemical deposition of an amorphous boron containing hydrogen film during helium glow discharge with 10% B_2H_6 [1]. The most important impact of the boronization was the strong decrease in low-Z impurities: oxygen and carbon.

The reduction of the impurity sources owing to the boronizations was studied using a set of spectroscopy systems installed on Wendelstein 7-X and observing primarily the graphite divertor. This includes in particular an overview spectrometer system [2] and a part of an endoscopic detection system [3] which allowed to measure emission spectra in a wide range from $300\,\mathrm{nm}$ to $1100\,\mathrm{nm}$. Such a wide wavelength span allowed the simultaneous observation of hydrogen recycling fluxes (Balmer lines), boron, carbon, and oxygen impurity fluxes by recording the line emission of neutral and low ionization stages such as CII, BII. To assess the homogeneity of the observations and the screening of impurities, complementary systems located at different toroidal positions and observing the edge and core plasmas and higher ionization stages were used.

A substantial reduction of the local oxygen and carbon particle and line emissions by a factor of 10 and 3 correspondingly was recorded after the first boronization by comparing reference discharges. Simultaneously, newly arisen boron line emissions were observed. In OP1.2b of Wendelstein 7-X, in total three boronizations were applied leading to a substantial reduction of the O levels in the edge and core plasma. Between two boronizations intensity of oxygen increased by a factor of 4 but has never reached the pre-boronization values. Carbon level increased and boron level

decreased (by a factor of 4) between the boronizations due to erosion. This behavior has been investigated after operation in different magnetic configurations of Wendelstein 7-X and touching different areas of the divertor.

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Chemical erosion measurements on the carbon divertor plates of W7-X

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Chemical sputtering via the formation and desorption of hydrocarbon molecules and physical sputtering by hydrogen and impurities are the main erosion mechanisms on the carbon plasma-facing components (PFCs) of fusion devices [1]. The chemical erosion is enhanced by the residual oxygen via forming carbon oxide molecules even at lower temperature [2]. In order to study the chemical erosion, hydrocarbon injection experiments have been widely performed in many fusion devices [3-6]. The carbon chemical erosion is quantified by the spectroscopic measurement of hydrocarbon break-up products CH and C_2. The discrimination of chemical sputtering from physical sputtering is accomplished by quantitatively relating the fraction of $C]I$ influx expected from dissociation of hydrocarbon to the measured CH and C_2 influxes. In W7-X, ten test divertor units (TDU) made of graphite, two in each period of the fivefold symmetric device, have been installed. The magnetic footprints on divertor plates are strongly depend on the basic magnetic configurations which mainly include stand ($\iota_a = m/n = 5/5$), low/high iota ($\iota_a = m/n = 5/6, 5/4$), low/high mirror ($\iota_a = m/n = 5/5$), and inward/outward shifted ($\iota_a = m/n = 5/5$) divertor configurations.

In OP1.2b, CH_4 injection through the thermal He-beam diagnostic installed on divertor plates was performed. The CH Gerö and C_2 swan band from hydrocarbon break-up products was applied to determine the in-situ hydrocarbon fluxes and quantify the chemical erosion. The chemical erosion on divertor targets in different magnetic configurations were compared. The enhanced chemical erosion from oxygen was quantified as well.

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Design of a microwave reflectometer diagnostic for the density profile measurement at the ICRF antenna in W7-X

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A fast sweep frequency modulated continuous wave (FMCW) reflectometer is designed for the electron density profile measurement at the ICRF antenna in Wendelstein 7-X (W7-X). This system covers the E and W frequency bands ($67.2-110\mathrm{GHz}$) in the right-hand X-mode polarization, which corresponds to a cut-off density range of ne $n_\mathrm{e} \leqslant 6 \times 10^{19}\mathrm{m}^{-3}$ for the central magnetic field $B_0 = 2.5\,\mathrm{T}$. The Ka-band waveguides (WR-28) are equipped in the vacuum, which length is $3.5\,\mathrm{m}$ per path. Due to a low space availability and minimization of the losses, the waveguides are banded with certain angles. Two pyramidal horns are placed between the rods of the ICRF's Faraday shield and the frame. The horns with an elongated H-plane are manufactured for a sufficient directivity and gain. In the reflectometer electronic, a layout for a fast frequency sweep in the heterodyne measurement is assembled. This scheme is similar to the other profile reflectometers presented on EAST[1] and Tore Supra[2] tokamaks. The reflectometer electronic is integrated into an individual metal box, which will be placed on the upper layer of the ICRF antenna sliding carrier.

This reflectometer is under calibration in the laboratory. Both the electronic and the transmission line are required to be fully tested before the installation to the ICRF system. According to the schedule, this reflectometer diagnostic will be operated in the OP 2 campaign of W7-X for the edge density profile measurements in front of the ICRF antenna. The underlying physics with respect to, e.g., wave particle coupling, SOL density behaviors, turbulent transport will be conducted.

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Impact of magnetic configuration on the ion temperature profile in W7-X plasma boundary by a retarding field analyzer probe

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Understanding the impact of magnetic configuration on the ion temperature (T_i) profile at the plasma boundary is critical for avoiding damage of the plasma-facing components in the high beta steady-state operation on Wendelstein 7-X (W7-X). Here, the T_i profile at the plasma boundary has been successfully measured on W7-X during the last experimental campaign OP1.2b by a retarding field analyzer (RFA) probe, which was previously commissioned at EAST[1]. The RFA probe head was mounted on the front-end of a multi-purpose manipulate [2] located at the outer midplane of W7-X. The experimental observations show that the ion temperature profile location is $\approx 5\,\mathrm{cm}$ radially inward when the magnetic configuration changed from the standard configuration (EIM+252) to the high iota configuration (FTM+252). In the EIM+252 configuration, an ion temperature shoulder has been observed near the location where a sudden change of the field line connection length appears in the scrape-off-larger (SOL) region, while no shoulder was observed for plasma with the FTM+252 configuration, and the gradient of T_i increases due to narrower island width.

Furthermore, the edge ion temperature profile has been studied by slightly varying the edge island size by turning the control coil current Icc, ECRH heating power PECRH and plasma density in the EIM+252 configuration. The results show that the ion temperature measured at the edge shoulder region (T_is) decreased when the island width was expanded. For a fixed control coil setting, Tis increases gradually with the increase of the ECRH heating power. A decrease of T_i was also observed when the central line-integrated electron density, nel , increases from $7 \times 10^{19}\mathrm{m}^{-2}$ to $9 \times 10^{19}\mathrm{m}^{-2}$.

Nitrogen seeding from the upper horizontal divertor port M51 was performed on W7-X, a significant reduction and flattening in the edge ion temperature profile has been observed. In addition, the edge ion-to-electron temperature ratio has been obtained. Here, the electron temperature is measured by a combined probe[3]. The calculated is close to 1 in the far SOL region and then increases to ≈ 3 as the probe moved into the shoulder region. These observations could help to guide the control of upstream ion temperature in the future W7-X steady-state experiments.

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Effects of HFS n=1 error field and its LFS correction on H-mode and L-H transition on COMPASS

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Experiments with HFS-generated error fields (EF) have recently been performed on the COMPASS tokamak, exploiting its external magnetic perturbation coils with the unique poloidal coverage of the vessel including HFS. The inboard-generated EF are of crucial importance for the strategy of the EF correction at ITER, given that a considerable error field might occur e.g. due to the tilt of the central solenoid, and that the impact of these EF might be qualitatively different from the impact of EF generated on LFS, which was the only one studied in past experiments. Simulations of the ideal MHD code IPEC have shown that generally in low beta plasmas the coupling of HFS EF to individual rational surfaces (e.g. 2/1, 3/1, 4/1) does not happen with the same toroidal phase, unlike for LFS EF [1], thus complicating the compensation of HFS EF by LFS error field correction scheme.

In this work we document the complications associated with the presence of uncompensated HFS EF (e.g. due to CS tilt) during the L-H transition as observed on the COMPASS tokamak. Specifically, the prevention of the L-H transition in low beta plasmas and possible H-L back-transition in higher beta plasmas are demonstrated. We report that the observed detrimental effects may be alleviated by using an appropriate correction field from LFS, preventing the H-L back-transitions and reducing the occurrence of disruptions during L-H transition by up to 75% (depending on the configuration of the correction field). Modelling by IPEC implies that the beneficial effect of the applied correction field is mostly due to the compensation of the 2/1 core resonant component.

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Theoretical understanding on error field penetration in EAST

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One of the great challenges for the steady state operation in tokamak plasma is the error field problem. Error field can induce locked mode and result in major disruption has been recognized since 30 years ago. According to the theory extrapolation, the tolerance of error field is smaller for a larger device. Therefore, error field has long been a concern in many famous tokamaks. However, theoretical analysis seldom gives satisfied answers for the error field scaling experiments.

To better understand the error field problem, series of error field penetration experiments using n=1 and n=2 RMP have been carried out in EAST [1,2,4,5]. To understand the spectrum effects, the MARS-F code is employed [1,2]. As one of the essential issues in error field study, density scaling on field penetration is also a significant concern. A similar density scaling has been observed using different n=1 RMP spectrum [1]. One of the most uncertainties in determining the density scaling is the density dependence of momentum dissipation time, which significantly influences the final scaling [3]. The density dependence of momentum dissipation time is estimated from energy confinement time measured in the experiment. It is almost independent of plasma density. This means that the usually used assumption of Neo-Alcator linear scaling is not applicable here. The obtained theoretical scaling on the basis of experiment momentum dissipation time successfully explained the observed experimental scaling [1].

Moreover, the very recent results on toroidal field scaling of n=1 RMP mode penetration and n=2 RMP mode penetration accompanied by the linear response analysis are also given. Furthermore, are given. In addition, some other new observations on n=2 error field penetration, e.g. field penetration scaling, excitation of n=2 TAE mode by barely trapped energetic electrons accelerated by force reconnected island etc., are also presented [6].

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Variability of Magnetic Configuration Properties in Wendelstein 7-X

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The stellarator experiment Wendelstein 7-X (5 field periods, major radius of $5.5\,\mathrm{m}$ and minor radius of $0.5\,\mathrm{m}$), which went into operation in December 2015, has been optimized with respect to various properties, whose improvement is mandatory for stellarator fusion reactors: MHD-equilibrium and - stability, low neoclassical transport in the long-mean-free-path regime as well as low bootstrap current and good fast particle confinement. The optimized theoretical configuration is realized by a coil system of 50 non-planar coils (5 coil types with independent power supplies) and 20 planar coils (2 coil types), allowing for a considerable flexibility in tailoring magnetic configurations. Since one aim of the operation of Wendelstein 7-X is to show quasi-steady-state operation at fusionrelevant plasma parameters, the device has been equipped with superconducting coils and with an island divertor for power and particle exhaust, which relies on the presence of a low order rational island chain at the proper radial location between the divertor structures and the main plasma.

This contribution describes the properties of the magnetic configurations of Wendelstein 7-X with respect to variations in the coil currents, with special attention on the optimization goals. Furthermore, the influence of the plasma parameters on the configuration properties is discussed, which includes the effect of plasma currents on island divertor operation.

Plasma contributions to the edge magnetic field of Wendelstein 7-X

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The Wendelstein 7-X (W7-X) experiment has just concluded operation of its second divertor campaign. In this contribution, we present measurements of the edge magnetic field profile using the multi-purpose manipulator [1][2] and the combined probe [3]. These measurements are performed to provide validation for MHD equilibrium models that will be used to investigate high-beta scenarios for the planned high-performance campaign with a cooled divertor.

Measurements with this probe were performed during various experiments (with varying conditions such as magnetic configuration, ECRH power, electron density and plasma beta), including long-running discharges of up to 100 s duration. The plasma-induced changes in the magnetic field are being correlated to plasma beta and toroidal current measurements.

For this campaign, the combined probe was upgraded with a chopper-stabilized analog hardware integrator, working in tandem with raw coil signal acquisition. This hardware integrator provides a signal with (compared to post-acquisition software integration) reduced low-frequency noise. Preliminary results indicate that the sensitivity is sufficient to detect magnetic field changes even at lower beta, but the full characterization of the system is still in progress.

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Influence of resonant magnetic perturbations on the divertor power distribution based on modeling of the diffusion-combined three-dimensional magnetic topology

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The success of edge localized mode (ELM) suppression by resonant magnetic perturbations (RMPs) provides a promising method to eliminate the large transient power flux to the plasma facing components (PFCs). The edge stochastic region induced by RMPs has been confirmed from experiments that it can affects the edge plasma transport and the power load distribution on divertor targets [1].

To better understand the physical mechanism behind, a field line diffusion model is implemented in the 3D magnetic topology code TOP2D [2] for the prediction of the divertor power load distribution. It assumes a simplified SOL model with the free-streaming parallel heat convection and with anomalous cross-field diffusion transport. The diffusion model is achieved by simulating a random walk process with the Monte-Carlo method to determine the distributions of the collision frequency and radial displacement. The modeling results in EAST show that the diffusion process increases the connection length and penetration depth of field lines connected to divertor targets. The shapes of strike point splitting on targets are not much changed but the power deposition ratio between the original and additional strike points changes with different diffusion coefficients. Both the increased diffusion coefficient and amplifying effects of plasma responses can increase the portions of total input power that deposit on the targets connected to the second separatrix.

This simplified model can make a rough estimation of the power distribution on the PFCs and largely reduce the computation time and resource as compared to the calculation by EMC3-EIRENE code [3, 4], especially for the calculation with large amount of cases for power load predictions. More detailed physical analyses should be dealt with by a more comprehensive and self-consistent model like the code mentioned above.

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Threshold and dynamic process of mode excitation by rotating m/n=2/1 resonant magnetic perturbation in J-TEXT tokamak

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Experiments of rotating field penetration have been carried out in J-TEXT recently. By applying the rotating resonant magnetic perturbation (RMP) with a dominating mode numer m/n=2/1, a rotating 2/1 tearing mode is forced to form and locked in frame of the RMP. Compared with the penetration of static RMP, the rotating RMP penetration is of great benefits for the observation and measurement of the field penetration process. Weak magnetic perturbations of the plasma response have been observed at the 2/1 screen stage, which have a 90 degree phase difference with the RMP. This phase difference turn to 0 after a rotating mode has been excited. From the space distribution of the magnetic responce, a 3/1 mode structure was found in the 2/1 screening stage and the ealry 2/1 penetration stage. That might indicate the 3/1 mode play a role in the 2/1 field penetration. The influence of rotating RMP penetration on the plasma parameters are also investigated comparing the statci RMP penetration [1].

In addition, the rotating RMPs with different frequencies were applied for varying slip frequencies in the experiments. Here, the slip frequency is the frequency difference between the RMP and the electron flow of the plasma. The RMPs have a maximum frequency at 5 kHz, while the frequency of the intrinsic 2/1 mode is around 7 kHz. The experimental results show that the field penetration threshold will increase with the increase of the edge edge safety factor and decrease with the increasing frequency of the RMP at the co-MHD direction. These results are in agreement with theory and the experimental results at TEXTOR, which varied the slip frequency via neutral beam injectors [2].

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Effects of resonant magnetic perturbations and induced islands on plasma flow and turbulence in J-TEXT Ohmic plasmas

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Recently, dedicated experiments have been carried out on J-TEXT tokamak for investigating edge plasma responses to the externally applied resonant magnetic perturbation (RMP) field with four different toroidal phases under 2/1 and 3/1 modes, by using the Langmuir probe and laser collective scattering diagnostics. Under the dynamic 2/1 RMP penetrating, the excited 2/1 rotating island has strong modulation effects on plasma density (n_e), temperature (T_e), floating potential (V_f), microfluctuation and perpendicular flux. The most interesting phenomenon is that, the V_f shows positive variation outside of the O-point of the island, and negative variation outside of the X-point. This experimental result has been confirmed by the simulation result of the EXFC code, which found that the island has stron effects on the ITG micro-instability and results in positive potential perturbation outside of the O-point and inside of the X-point. Additionally, the laser collective scattering spectrum shows that the Trapped Electron Mode (TEM) located inside the 2/1 resonant surface enhances obviously after the 2/1 island appears.

Under the static 3/1 RMP penetrating, the Langmuir probe could measure across the excited static 3/1 island and identify the radial position of the island. Then, the influences of the 3/1 island on edge plasma flows, turbulence and flux has been investigated. It is found that the poloidal mean flow near the island changes from electron to ion diamagnetic direction after RMP penetration. The toroidal velocity changes largely inside the X-point and contains near the O-point.The entire turbulence is depressed near the O-point, while the low frequency turbulence ($< 50\,\mathrm{kHz}$) enhances near the X-point. In addition, the Geodesic Acoustic Mode (GAM) is depressed after the appearance of the 3/1 island. The variance of turbulence is explained by the change of poloidal flow. This experimental analysis is continued and more results will be presented on the conference.

Active controlling of the divertor flux by the supersonic-molecular-beam-injection with LHW-induced magnetic perturbations on EAST

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Experiments on EAST indicate that lower hybrid waves (LHWs) can profoundly change magnetic topology by inducing helical current filaments flowing along magnetic field lines in the scrape-off layer [1,2,3]. Recently, the redistribution of the divertor flux caused by the synergy of the supersonic-molecular-beam-injection (SMBI) and LHW-induced magnetic perturbations has been observed on EAST with the ITER-like tungsten divertor [4]. To reveal the physical mechanism behind, simulations with good qualitatively agreements are performed by utilizing the fluid 3D edge plasma Monte-Carlo code (EMC3) coupled to the kinetic neutral particle transport code (EIRENE)[5,6]. The ions and electrons caused by the ionization of injected neutral particles in the scrape-off layer flow along the magnetic flux tube towards of the divertor, thus directly increasing the heat and particle flux on the split strike lines. Combining this conclusion with the features of magnetic topology, actively controlling of the divertor flux can be realized by adjusting the SMBI position or the phase of magnetic perturbations.

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Optimized map projections with minimal distortion

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In the 19th century, Gauß proved that the surface of a sphere cannot be unfolded onto a flat plane without distortion, hence every map projection necessarily introduces at least some distortion, and no flat map of the Earth can be perfect. Later in the 19th century, Tissot presented his indicatrix as a means of providing a visual representation of area and shape distortions in map projections [1]. In the 21st century, attempts have been made to quantify distortions in map projections in order to produce rankings of least distorted projections. In [2], a ranking of map projections according to the fraction of the area of each map in which measures of distortion are below arbitrary threshold acceptable values is presented.

In [3], it is shown how area and shape distortions can be calculated from the metric tensor of a map projection, while flexion and skewness distortions, which are calculated from the first derivatives of the metric tensor, are introduced for the first time. Accordingly, the Goldberg–Gott indicatrix is presented as an updated version of the Tissot indicatrix and provides a visual representation of area, shape, flexion and skewness distortions. Finally, a ranking of map projections according to area, shape, distance, flexion, skewness and boundary distortions averaged over the surface of the globe is presented. The least distorted map projection according to this criterion is the Winkel Tripel projection, which has the unfortunate property of not having an analytical inverse projection. In other words, although each point on the map can be calculated analytically from given latitude and longitude coordinates on the globe, the latitude and longitude coordinates on the globe can only be calculated numerically from a given point on the map, and this numerical calculation is complicated.

Although the sophisticated method of quantifying distortions in map projections presented in [3] has been known for over a decade, no attempt has been made until now to produce new map projections optimized to have minimal distortion. Here, new map projections having less distortion than existing projections according to the criterion in [3] are presented. Furthermore, these map projections are not only analytical but also have an analytical inverse projection.

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The simulation of runaway current suppression by resonant magnetic perturbation in the plasma disruption on J-TEXT

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Simulations of runaway current suppression by resonant magnetic perturbation (RMP) in plasma disruption triggered by massive gas injection (MGI) on J-TEXT were performed with the 3D MHD code NIMROD. It was found that different phases and amplitudes of RMP can significantly affect the generation of runaway current during the plasma disruption triggered by Ar MGI. A test particle model was applied to trace the trajectory of the runaway electrons (REs). The simulation results indicate that RE confinement is affected drastically by the magnetic topology structure or magnetic perturbation during the thermal quench (TQ) phase. Furthermore, the dependence of the runaway confinement on different phases and amplitudes of m/n=2/1 dominated RMP was investigated. It was found that both different phases and amplitudes of RMP result in different values of the runaway confinement plateau during the current quench (CQ) phase. Different phases of RMP can result in different ways of impurity spreading and eventually lead to a discrepancy in the magnetic topology structure and magnetic perturbation amplitude. Different amplitudes of RMP can directly result in different average levels of magnetic perturbations to affect RE confinement during the TQ.

Minor disruptions triggered by supersonic molecular beam injection in the J-TEXT tokamak

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Non-linear magnetohydrodynamic (MHD) simulations of the argon (Ar) supersonic molecular beam injections (SMBI), that triggered disruptions in the J-TEXT tokamak, are performed with the 3D MHD code NIMROD and compared to experimental data in detail. The SMBI simulations reproduce experimentally observed phenomena such as the destabilization of MHD modes during the impurities cold front prenetration phase, followed by the collapse of the $q \geqslant 2$ region and a sequence of partial collapse in the core during minor disruptions. The mode coupling of m/n=2/1,3/1 and the core instability drives for successive impurity and heat transport events of the $q \leqslant 2$ region.

The 2/1 mode plays a crucial role in the disruption. The growth of the core instability is directly related to the magnetitude of the 2/1 mode. The interaction between the core instability and 2/1 mode determines the degree of disruption. The mechanism of minor disruption and major disruption are discussed.

Conceptual design of the island divertor configuration in the J-TEXT tokamak

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The high heat load on divertor target is one of the essential issues that must be addressed for future fusion reactors. In tokamak, it is found in the heuristic drift-based model [1] that power decay length λ_q is proportional to the connection length $L_\parallel$. In the scrape-off layer (SOL) of a tokamak, $L_\parallel$ is inversely proportional to the poloidal magnetic field Bp. Many improvements of poloidal divertor were made to reduce the heat load on target, e.g. flux expansion and radiation divertor. An alternative concept is using the island divertor (ID) configuration, which proposed in 1977 for tokamak [2]. It has been established successfully in tokamak, e.g. TEXTOR, and stellarators [3], such as W7-AS, LHD and W7-X. The ID configuration not only has a long L|| in the SOL, but also forms a three-dimensional stochastic layer around the boundary magnetic island. It is beneficial to increase the equivalent radial transport, and consequently reduce the peak heat load on the divertor target [4].

In the J-TEXT tokamak, the external resonant magnetic perturbation (RMP) coils are employed to generate the boundary magnetic island. The magnetic field line of this island is opened by divertor targets, so as to form the ID configuration. Previously, the m/n=3/1 island in the edge plasma has been formed successfully in J-TEXT. However, it is found that the 2/1 locked mode could be easily excited after the formation of a 3/1 locked island, especially at a low density and high RMP amplitude. In order to avoid exciting 2/1 locked mode, we select the 4/1 island. For an island divertor, a sufficient width of the boundary island is needed. But higher m and stronger magnetic shear may increase the difficulty to form a large boundary island. To form an island with critical width of approximately $2.5\,\mathrm{cm}$, the applied 4/1 RMP field needs to exceed $8\,\mathrm{G}$ in the edge for the typical boundary plasma parameters on J-TEXT.

Recently, numerical calculations have been carried out to calculate several parameters for the ID configuration such as the perturbed magnetic field, its spectrum, Poincare plot, connection length, striking point and so on. The ID configuration can be established by the boundary 4/1 island intersecting with proper divertor targets.The detailed design and other researches will be presented in the conference.

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Combined effects of stochastic magnetic fluctuations and synchrotron radiation on the production of runaway electrons

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The dynamics of relativistic runaway electrons are analyzed using the relativistic Fokker-Planck equation including deceleration due to the synchrotron radiation and radial diffusion loss caused by stochastic magnetic fluctuations (SMFs). SMFs are treated as friction force in [1]. However, we think SMFs act as a "porter" in configuration space, but not directly affect the runaway electrons (REs) in momentum space. Both critical electric fields for sustainment of the existing REs and for avalanche onset are enhanced, and the modified avalanche growth rate is reduced by the combined effects of SMFs and synchrotron radiation as compared to the case with only synchrotron radiation [2].

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Modelling plasma boundary corrugation and tailoring toroidal torque profiles due to application of 3D fields with ELM control in ITER

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Type-I edge localized modes (ELMs) can lead to a large amount of particles and energy being released from the plasma into the plasma facing components, which is potentially dangerous to the future tokamak devices, especially for ITER [1]. The external magnetic perturbation (RMP) technique has been successfully established as a relatively mature method for controlling type-I ELMs on several present-day fusion devices [2-6]. Previous work [7-9] has shown that the normal displacement of the plasma surface caused by RMP fields and computed by MARS-F [10], can be used as a reliable indicator to describe global features of the plasma response and the resulting transport properties that affect the behavior of ELMs.

In this work, based on the 3D plasma surface boundary displacement produced by the RMP fields, we investigate the plasma response for five representative ITER plasmas, designed for different phases of the ITER exploration. It is found that, the plasma response amplitude (normalized by the equilibrium plasma current) is largely dictated by q95. The response depends on the pedestal rotation as well as the applied toroidal spectrum.

Based on the computed plasma response, we also investigate the potential of influencing the plasma flow profiles by RMP, by optimizing the radial distribution of toroidal torques. Using the ITER scenario with 15 MA/5.3 T and fusion gain of Q=10 as the case study, we find a strong coupling effect between the core and edge torques, which is largely associated to the middle-row coil contribution. Utilizing the two off-middle rows of ELM control coils helps to decouple the core and edge torques.

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Excitation of locked islands by static and rotating resonant magnetic perturbations on J-TEXT

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The locked magnetic island in the core is one of the biggest threats to the tokamak plasma operation, since it can lead to major disruption. The threshold and dynamic process of the locked island excitation, due to the penetration of resonant magnetic perturbation (RMP), are investigated on J-TEXT by utilizing the flexible RMP system, which can provide either a static or a high frequency (up to $6\,\mathrm{kHz}$) rotating RMP field. The threshold of the locked island, excited due to the penetration of static 2/1 RMP, was found to increase with the increase of electron density (n_e), but dramatically decreased once n_e larger than a critical value. By modifying the toroidal rotating (V_Φ) using the biased electrode, the RMP penetration threshold increased linearly with V_Φ. The penetration of high frequency rotating RMP (RRMP) field were observed with reduced edge safety factor $q_\mathrm{a} < 2.8$. The excitation of this rotating island might be related to the kink mode with low initial slip frequency ($f_\mathrm{s} = f_\mathrm{MHD} - f_\mathrm{RMP}$) and low q_a, both of which influenced the threshold of RRMP penetration significantly.

The locked island in the edge, in contrast to the core locked island, can be helpful for the control of edge instabilities and plasma wall interaction. The 3D boundary due to the formation of static magnetic island was established in the edge via two methods, (a) by the penetration of RMP with dominant 3/1 component at $q_\mathrm{a} = 3$; (b) by forming the non-axisymmetric helical current (NAHC) in the SOL due to a biased electrode. The SOL NAHC was confirmed by the magnetic measurement and they can produce MP field resonant with the edge rational surfaces. Experimental evidences indicate the formation of edge island due to the NAHC.

Numerical error estimation in EMC3-EIRENE for a DIII-D divertor edge plasma

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The iteratively solved Monte Carlo (MC) code EMC3-EIRENE [1] is frequently used for plasma edge simulation in 3D applications. Currently, however, the numerical accuracy is assessed in a very limited way. Typically, convergence is monitored via the relative change of the simulation result over subsequent iterations. This relative change is often considered to be a measure for the statistical error. It has been demonstrated, however, that this is insufficient as an error estimate since the relative change can give a misleading indication and neglects deterministic error contributions [2]. Deterministic errors arise from solving non-linear problems with a finite number of MC particles (finite sampling bias), advancing fluid Monte Carlo particles with a discrete time step (time integration error), and using discretized variables in space (discretization error). These error contributions can be estimated based on the error reduction rates by comparing solutions with a different resolution. To obtain reliable estimates, it is important to have a sufficiently low variance on the solutions, which can be efficiently obtained by using post-processing averaging. This does not only enable accuracy analysis, but can also significantly reduce the required CPU time.

In this work, we analyze the numerical accuracy in an EMC3-EIRENE simulation of an axisymmetric DIII-D divertor edge plasma. As a first step, we focus on the estimation of the error contributions on a specific numerical grid, excluding the discretization error. By varying the time step and the number of MC particles per iteration, the expected error reduction rates are confirmed and the magnitude of the errors is estimated. Using parametrized expressions for the error and the computational time, optimal numerical parameters can be determined to achieve faster and/or more accurate results.

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Measurement of the divertor heat flux distribution using the infrared camera in the EAST plasma with lower hybrid current drive

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During the discharge of the Experimental Advanced Superconducting Tokamak (EAST), the divertor target plate (DTP) accumulates a large amount of heat and suffers irreversible damages, so building of a temperature measurement system and analyzing the heat flux distribution are crucial for guaranteeing the steady-state operation. In this work, a temperature measurement method using a proposed nonlinear emissivity is developed to obtain more accurate heat flux results. Firstly, the infrared (IR) camera diagnostic system is calibrated with a standard blackbody furnace using a multi-temperature method, which effectively eliminates the effect of the IR system on the temperature measurements. Based on the theory that emissivity changes with temperature, the nonlinear emissivity model is built to more accurately determine the temperature rather than using a supposed constant emissivity. With the measured surface temperature, the heat flux can be calculated by resolving heat transfer equations [1].

Previous experimental results from EAST have shown that the lower hybrid current drive (LHCD) can induce a 3D magnetic topology change at the plasma boundary, thus affect significantly the divertor heat flux distribution [2, 3]. In this paper, the effects of LHCD on the heat flux distribution will be discussed. The experimental observation shows multiple peaks appearing in the heat flux profile along the divertor plate, which is considered to be due to the splitting of the strike line during the application of LHCD.

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Simulations on W impurity transport in the edge of EAST H-mode plasmas

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This paper provides an investigation of the production/transport properties of W impurity in the edge of EAST H-mode discharges with upper-single-null (USN) configurations by using DIVIMP Monte-Carlo code. The background plasmas are provided by SOLPS5.0 calculations. Firstly, to address the detailed dependence of W impurity behaviors on plasma conditions in the SOL/divertor region of EAST, two pre-ELM cases with divertor plasmas in high-recycling and partially-detached regimes, respectively, are considered and compared. It is found that, due to the competitions between the thermal and friction forces, in the high-recycling case, large quantities of W impurities are transported to the upstream; while in the partially-detached case, most of the W impurities are located near the inner and outer divertors. Furthermore, the W core contamination in type-I ELMy H-mode plasmas of EAST has been estimated by the DIVIMP-SOLPS5.0 simulations, without the consideration of the release of W impurity from the core due to ELMs. During the ELM, simulations exhibit the low-recycling regime of divertor plasma, and high (several hundred eV) target plasma temperatures together with low SOL collisionality ($\eta^*_{\rm sol} \approx 1-3$), lead to an order of magnitude increase in the W core contamination rate. However, in the ELM-recovering phase, with the decrease (increase) of divertor plasma temperature (density), significant increase of divertor retention is obtained due to the remarkable increase of friction forces on W ions. Besides, intra-ELM phase approximately contributes to more than 50% of the total W core contamination per an ELM cycle. This work represents a step toward a deeper understanding of the W impurity production/transport in EAST H-mode discharges.

Optical reflection coefficient measurements for the identification of deposition layer in LHD and W7-X

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A method for simply and extensively obtaining the thickness information of the deposition layer is provided by employing an innovative concept of optical reflection coefficient measurement by colour analyzer in LHD [1]. In the colorimetry, the wide range of distribution of deposition layer was obtained. As a result of the analysis, a deeper discussion of the quantitative fuel retention into the wall is possible. The colorimetry of the plasma facing components in W7-X has been also performed after the experiment campaign with graphite divertor operation in Operation Phase (OP) 1.2a and 1.2b. We will present the optical reflection coefficient measurement between LHD and W7-X.

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Influence of the plasma current on the edge profiles at W7-X

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Wendelstein 7-X (W7-X) has recently finished the second operational campaign (OP 1.2), which featured an island divertor. The milestone achievements were a central density in the order of 1020 m-3 and a pulse durations of up to 100 seconds (in separate discharges).

Both the high performance plasma and the wide range of configurations and tuning of the control coils made a study of configuration effects and the influence of the plasma on the magnetic topology possible. This was especially the case for the "standard configuration" where the 5/5 island remnant was observed with a varying position and size. Due to the high performance and plasma beta the discharges also experienced a dynamic current evolution, leading to toroidal currents exceeding 10 kA.

The multipurpose manipulator [1], in the mid plane of W7-X, was used together with a variety of fast reciprocating probes [2] [3] to measure the electron temperature, density, the radial electric field and local magnetic field in the edge region. The remnant island can be identified with the profiles of, for example, the floating potential, the electron temperature and the electric field.

These modifications are documented by the use the field line tracing web service with modified configurations that show a strong shifting of the island remnant and the connection length profiles. The EMC3-EIRENE modelling [4] was used with additional data from diagnostics such as the divertor Langmuir probes and the Bolometry to confirm the measured profiles.

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Scenario with Combined Density and Heating Control to Reduce the Impact of Bootstrap Current in Wendelstein 7-X

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Wendelstein 7-X (W7-X) is a low-shear stellarator with 10 modular island divertor units designed for particle and heat exhaust. For the island divertor concept to work properly, the device is optimized for small internal currents, in particular, the bootstrap current (BSC) is minimized. Previous studies have predicted an overload situation near the pumping gap of the divertor that occurs at full heating power at about half-way of the evolution of the net toroidal current.[1]

The present numerical study, in the framework of neoclassical theory, and assuming certain amount of anomalous transport, demonstrates that a self-consistent path from low density and low heating power to high density and full heating power exists, on which the BSC remains constant as long as both quantities are varied in a coordinated way. The values of density in the path for each power step proposed are below empirical density limit of W7-X. [3]

For this study, the numerical tools used are the Variational Moments Equilibrium Code (VMEC) and the EXTENDER-code to calculate the magneto-hydrodynamic equilibrium field produced by the plasma and the external coils in the entire vacuum chamber. The plasma currents are calculated self-consistently by iterating between various codes until changes are negligible, as follows: the BSC is based on a VMEC-equilibrium calculation that serves as an input to calculate the transport coefficients with the Drift-Kinetic Equation Solver (DKES). These are used in transport simulations [2] to predict the plasma profiles, which then provide input for the next equilibrium calculation with VMEC. Based on the sequence of equilibrium fields from these self-consistent scenarios, the compatibility of the thermal load to the targets and other wall components is examined using the field line diffusion code.

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